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LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

1984 ANNUAL REPORT
(OCTOBER 1, 1983 – SEPTEMBER 30, 1984)

ORIGINAL

Hanford Engineering Development Laboratory

Prepared by
W.N. McElroy (HEDL)
F.B.K. Kam (ORNL)
J.A. Grundl and E.D. McGarry (NBS)
A. Fabry (CEN/SCK)

Prepared for the U.S. Nuclear Regulatory Commission

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LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

1984 ANNUAL REPORT
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Hanford Engineering Development Laboratory

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P.O. Box 1970 Richland, WA 99352
A Subsidiary of Westinghouse Electric Corporation

Prepared by
W.N. McElroy (HEDL)
F.B.K. Kam (ORNL)
J.A. Grundl and E.D. McGarry (NBS)
A. Fabry (CEN/SCK)

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FOREWORD

The Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) has been established by NRC to improve, test, verify, and standardize the physics-dosimetry-metallurgy, damage correlation, and associated reactor analysis methods, procedures and data used to predict the integrated effect of neutron exposure to LWR pressure vessels and their support structures. A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the US NRC-supported activities at HEDL, ORNL, NBS, and MEA and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Jülich, Germany), and several UK laboratories have been extended to a number of other countries and laboratories. These cooperative links are strengthened by the active membership of the scientific staff from many participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR surveillance standards.

The primary objective of this multilaboratory program is to prepare an updated and improved set of physics-dosimetry-metallurgy, damage correlation, and associated reactor analysis ASTM standards for LWR pressure vessel and support structure irradiation surveillance programs. Supporting this objective are a series of analytical and experimental validation and calibration studies in "Standard, Reference, and Controlled Environment Benchmark Fields," research reactor "Test Regions," and operating power reactor "Surveillance Positions."

These studies will establish and certify the precision and accuracy of the measurement and predictive methods recommended in the ASTM Standards and used for the assessment and control of the present and end-of-life (EOL) condition of pressure vessel and support structure steels. Consistent and accurate measurement and data analysis techniques and methods, therefore, will be developed, tested and verified along with guidelines for required neutron field calculations used to correlate changes in material properties with the characteristics of the neutron radiation field. Application of established ASTM standards is expected to permit the reporting of measured materials property changes and neutron exposures to an accuracy and precision within bounds of 10 to 30%, depending on the measured metallurgical variable and neutron environment.

The assessment of the radiation-induced degradation of material properties in a power reactor requires accurate definition of the neutron field from the outer region of the reactor core to the outer boundaries of the pressure vessel. The accuracy of measurements on neutron flux and spectrum is associated with two distinct components of LWR irradiation surveillance procedures 1) proper application of calculational estimates of the neutron exposure at in- and ex-vessel surveillance positions, various locations in the vessel wall and ex-vessel support structures, and 2) understanding the relationship between material property changes in reactor vessels and their support structures, and in metallurgical test specimens irradiated in test reactors and at accelerated neutron flux positions in operating power reactors.

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The first component requires verification and calibration experiments in a variet, of neutron irradiation test facilities including LWR-PV mockups, power reactor surveillance positions, and related benchmark neutron fields. The benchmarks serve as a permanent reference measurement for neutron flux and fluence detection techniques, which are continually under development and widely applied by laboratories with different levels of capability. The second component requires a serious extrapolation of an observed neutron-induced mechanical property change from research reactor "Test Regions" and operating power reactor "Surveillance Positions" to locations inside the body of the pressure vessel wall and to ex-vessel support structures. The neutron flux at the vessel inner wall is up to one order of magnitude lower than at surveillance specimen positions and up to two orders of magnitude lower than for test reactor positions. At the vessel outer wall, the neutron flux is one order of magnitude or more lower than at the vessel inner wall. Further, the neutron spectra at, within, and leaving the vessel are substantially different.

To meet reactor pressure vessel radiation monitoring requirements, a variety of neutron flux and fluence detectors are employed, most of which are passive. Each detector must be validated for application to the higher flux and harder neutron spectrum of the research reactor "Test Region" and to the lower flux and degraded neutron spectrum at "Surveillance Positions." Required detectors must respond to neutrons of various energies so that multigroup spectra can be determined with accuracy sufficient for adequate damage response estimates. Detectors being used, developed, and tested for the program include radiometric (RM) sensors, helium accumulation fluence monitor (HAFM) sensors, solid state track recorder (SSTR) sensors, and damage monitor (DM) sensors.

The necessity for pressure vessel mockup facilities for physics-dosimetry investigations and for irradiation of metallurgical specimens was recognized early in the formation of the NRC program. Experimental studies associated with high- and low-flux versions of a pressurized water reactor (PWR) pressure vessel mockup are in progress in the US, Belgium, France, and United Kingdom. The US low-flux version is known as the ORNL Poolside Critical Assembly (PCA) and the high-flux version is known as the Oak Ridge Research Reactor (ORR) Poolside Facility (PSF), both located at Oak Ridge, Tennessee. As specialized benchmarks, these facilities provide well-characterized neutron environments where active and passive neutron dosimetry, various types of LWR-PV and support structure neutron field calculations, and temperature-controlled metallurgical specimen exposures are brought together.

The two key low-flux pressure vessel mockups in Europe are known as the Mol-Belgium-VENUS and Winfrith-United Kingdom-NESDIP facilities. The VENUS Facility is being used for PWR core source and azimuthal lead factor studies, while NESDIP is being used for PWR cavity and azimuthal lead factor studies. A third and important low-fluence pressure vessel mockup in Europe is identified with a French PV-simulator at the periphery of the Triton reactor. It served as the irradiation facility for the DOMPAC dosimetry experiment for studying surveillance capsule perturbations and through-PV-wall radial fluence and damage profiles (gradients) for PWRs of the Fessenheim 1 type.

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Results of measurement and calculational strategies outlined here will be made available for use by the nuclear industry as ASTM standards. Federal Regulations 10 CFR 50 (Cf83) already requires adherence to several ASTM standards that establish a surveillance program for each power reactor and incorporate metallurgical specimens, physics-dosimetry flux-fluence monitors, and neutron field evaluation. Revised and new standards in preparation will be carefully updated, flexible, and, above all, consistent.

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CONTRIBUTORS LWR-PV-SDIP 1984 ANNUAL REPORT

Prepared by W. N. McElroy, F. B. K. Kam, J. A. Grundl, E. D. McGarry, and A. Fabry with reference to and/or use of selected contributions from the following participants and laboratories:

W. N. McElroy, L. D. Blackburn, R. Gold, G. L. Guthrie,
L. S. Kellogg, E. P. Lippincott, W. Y. Matsumoto,
J. P. McNeese, C. C. Preston, J. H. Roberts, F. H. Ruddy,
J. M. Ruggles, and R. L. Simons
Hanford Engineering Development Laboratory (HEDL), USA

E. D. McGarry, C. M. Eisenhauer, D. M. Gilliam,
J. A. Grundl, and G. P. Lamaze,
National Bureau of Standards (NBS), USA

F. B. K. Kam, C. A. Baldwin, R. E. Maerker, F. W. Stallmann, and M. Williams
Oak Ridge National Laboratory (ORNL), USA

A. Fabry, J. Debrue, G. DeLeeuw, S. DeLeeuw, P. J. D'Hondt, J. Lacroix,
G. Minsart, J. Pelsmaekers, H. Tourwé, and Ph. Van Asbroeck
Centre d'Etudes de l'Energie Nucleaire,
Studiecentrum voor Kernenergie (CEN/SCK), Mol, Belgium

W. Schneider, D. Packur, and L. Weise
Kernforschungsanlage (KFA) Jülich, Federal Republic of Germany

M. Austin, A. F. Thomas, R. Squires, and T. J. Williams
Rolls-Royce & Associates Limited (RR&A), Derby, UK

A. J. Fudge (Harwell) and J. Butler, I. J. Curl, P. C. Miller,
and A. Packwood (Winfrith)
Atomic Energy Research Establishment (AERE), UK

A. A. Alberman, J. Bollen, J. P. Genthon, and P. Soulat (Saclay)
and P. Mas (Grenoble)
Commissariat a l'Energie Atomique,
Centre d'Etudes Nucleaires (CEA/CEN), France

H. Farrar IV and B. L. Oliver
Rockwell International (RI), USA

G. R. Odette and G. Lucas
University of California at Santa Barbara (UCSB), USA

P. D. Hedgecock
APTECH Engineering Services, USA

J. S. Perrin and R. A. Wullaert
Fracture Control Corporation (FCC), USA

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B. A. Magurno and J. F. Carew
Brookhaven National Laboratory (BNL), USA

T. U. Marston, T. Griesbach, O. Ozer, T. Passell, and R. Shaw
Electric Power Research Institute (EPRI), USA

C. O. Cogburn, L. West, and J. Williams
University of Arkansas (UA), USA

W. C. Hopkins
Bechtel Power Corporation, USA

M. Haas
Engineering Services Associates (ENSA), USA

J. R. Hawthorne
Materials Engineering Associates (MEA), USA

E. B. Norris
Southwest Research Institute (SwRI), USA

M. P. Manahan
Battelle Memorial Institute (BMI) Columbus Laboratory, USA

S. L. Anderson, T. R. Mager, and S. E. Yanichko
Westinghouse Electric Corporation (W), USA

A. A. Lowe Jr, Q. King, R. H. Lewis, and C. L. Whitmarsh
Babcock & Wilcox Company (B&W), USA

G. C. Martin Jr
General Electric Company (GE), USA

G. P. Cavanaugh, S. T. Byrnes, and J. J. Koziol
Combustion Engineering, Inc. (CE), USA

N. Tsoulfanidis and D. R. Edwards
University of Missouri, Rolla (UMR), USA

S. Grant
Carolina Power and Light Company, USA

J. B. Sun
Florida Power and Light Company, USA

H. F. Jones
Maine Yankee Atomic Power Company, USA

F. Hegedus
Swiss Federal Institute for Reactor Research (EIR),
Wuerenlingen, Switzerland

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LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

1984 ANNUAL REPORT

W. N. McElroy (HEDL)

ABSTRACT

This report describes progress made in the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) during FY 1984. The primary concern of this program is to improve, test, verify, and standardize the physics-dosimetry-metallurgy and the associated reactor and damage analysis procedures and data used for predicting the integrated effects of neutron exposure to LWR pressure vessels and support structures. These procedures and data are being recommended in a new and updated set of ASTM standards being prepared, tested, and verified by program participants. These standards, together with parts of the US Code of Federal Regulations and ASME codes, are needed and used for the assessment and control of the condition of LWR pressure vessels and support structures during the 30- to 60-year lifetime of a nuclear power plant.

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The success of the LWR Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) continues to depend on the efforts and the free exchange of ideas and views by representatives of a large number of research, service, regulatory, vendor, architect/engineer, and utility organizations. The information reported herein could not have been developed without the continuing support of the respective funding organizations and their management and technical staffs. Special acknowledgment is due to C. Z. Serpan of NRC for having identified the need for an international program such as the LWR-PV-SDIP and for making it possible by taking a strong overall support and management lead.

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ACRONYMS

ASTM	American Society for Testing and Materials
BSR	Bulk Shielding Reactor
BWR	Boiling Water Reactor
CFS	Cavity Fission Source
DM	Damage Monitor
EOL	End-of-Life
HAFM	Helium Accumulation Fluence Monitor
HOTS	Hanford Optical Track Scanner
HSST	Heavy-Section Steel Technology (Program)
ISNF	Intermediate-Energy Standard Neutron Field
LWR	Light Water Reactor
NDTT	Nil Ductility Transition Temperature
NESDIP	Nestor Shielding and Dosimetry Improvement Program
NRC	Nuclear Regulatory Commission
NRE	Nuclear Research Emulsion
NSTF	Nuclear Science and Technology Facilities
ORR	Oak Ridge Research Reactor (ORNL)
PCA	Poolside Critical Assembly (ORNL)
PF	Perturbation Factor
PSF	Poolside Facility (ORNL)
PTS	Pressurized Thermal Shock
PuD	Paired-Uranium Detectors
PV	Pressure Vessel
PWR	Pressurized Water Reactor
RM	Radiometric Monitor
SDIP	Surveillance Dosimetry Improvement Program
SDMF	Simulated Dosimetry Measurement Facility
SPVC	Simulated Pressure Vessel Capsule
SRM	Special Research Material
SSC	Simulated Surveillance Capsule
SSTR	Solid-State Track Recorder
SUNY	State University of New York

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ACRONYMS* (Cont'd)

SVBC	Simulated Void Box Capsule
T/F	Thermal-to-Fast (Ratio)
TLD	Thermoluminescent Dosimeter
dpa	Displacement per Atom
TS	Thermal Shield
UK	United Kingdom
VENUS	Low-Flux Pressure Vessel Mockup, Mol, Belguim
WRSR	Water Reactor Safety Research

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LIGHT WATER REACTOR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

ANNUAL REPORT

1.0 INTRODUCTION

The objective of this program is to make measurements in neutron fields ("Benchmark" and reactor "Test and Surveillance Regions") for the subsequent validation/calibration of available state-of-the-art physics-dosimetry-metallurgy, damage correlation, and the associated reactor analysis procedures and data. These procedures and data are in turn used for predicting the integrated effects of neutron exposure to Light Water Reactor (LWR) Pressure Vessel (PV) and support structure steels from results of research reactor tests and power reactor surveillance programs. The program work includes: 1) selection of the neutron fields, 2) validation/calibration of physics-dosimetry-metallurgy, damage correlation, and the associated reactor analysis procedures and data using these fields, 3) preparation and editing of a series of 22 supporting NUREG reports, and 4) preparation and establishment of a set of 21 ASTM-recommended standard Practices, Guides, and Methods (see Figures 1 and 2).

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2.0 SUMMARY OF FY 1984 RESEARCH PROGRESS

To account for neutron radiation damage in setting pressure-temperature limits and making fracture analysis (see appropriate references in Section 4.0), neutron-induced changes in reactor PV steel fracture toughness and embrittlement must be predicted, then checked by extrapolation of surveillance program data during the vessel's service life. Uncertainties in the predicting methodology can be significant. The main variables of concern are associated with:

- Steel chemical composition and microstructure
- Steel irradiation temperature
- Power plant configurations and dimensions - core edge to surveillance to vessel wall to support structure positions
- Core power distribution
- Reactor operating history
- Reactor physics computations
- Selection of neutron exposure units
- Dosimetry measurements
- Neutron spectral effects
- Neutron dose rate effects.

Variables associated with the physical measurements of PV steel property changes are not considered here and are addressed separately in Appendices G and H of 10 CFR Part 50 (Cf83), in ASTM Standards, and appropriate references in Section 4.0.

The US NRC has estimated that without remedial action, there are a number of operating early-generation US pressurized water reactors (PWR) that could have beltline materials with marginal toughness, relative to the existing requirements of Appendices G and H and Regulatory Guide 1.99 (Re77), sometime within their presently licensed service life (Nr80,Ra83,Ra84); i.e., in the range up to about 32 years. This is of particular concern for safety, licensing, and regulatory issues related to pressurized thermal shock (PTS) (Di82).

As older vessels become more highly irradiated, the predictive capability for changes in fracture toughness and embrittlement must improve, particularly for plants operated beyond their current design service life; i.e., in the range above about 32 years. Since during the vessel's service life an increasing amount of information will be available from research reactor tests and power reactor surveillance programs, better procedures to evaluate and use this information can and must be developed. The most appropriate way to make information available on these procedures is through voluntary consensus standards, such as those now being developed by ASTM Committee E10 on Nuclear Technology and Applications (As84,He82,Mc84).

Important summary highlights of FY 1984 research activities of this multilaboratory program are:

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- The completion of the first draft and/or revisions of eight (Figures 1 and 2) of the 21 ASTM standards that focus on the physics-dosimetry-metallurgy, damage correlation, and the associated reactor analysis and interpretation aspects of the problem of guaranteeing the safety and integrity of the PV boundary and its support structures for LWR power reactors (As84,Mc84).

- The initiation and completion of important supporting verification and calibration benchmark studies, reviews, as well as neutron and gamma field experimental and calculational work (Table 1), which demonstrate and verify the direct applicability of the recommended methods, procedures, and data in the 21 ASTM standards (1 "master matrix," 9 "practices," 6 "guides," and 5 "methods"). Of particular interest here was the continuation of studies on: 1) the effect of the use of different neutron exposure parameters, 2) flux level and other variables effects, and 3) their separate and combined impact relative to the assessment and control of the present and end-of-life (EOL) condition of PV and support structure steels (Au83,Ch82,Ch83,Di82,Gu82,Gu82a,Gu84b,Gu84c,Gu84d,Mc84, Mc84h, Nr82,Ra83,Ra84,Si84). Of further interest was the continued planning and implementation of verification tests in H. B. Robinson, Maine Yankee, Crystal River (or Davis Besse), Arkansas-1 and Arkansas-2, see Figure 6, Table 2, and Ref (Mc84).

- The completion of the analysis of key experimental physics-dosimetry studies associated with the ORNL-Poolside Critical Assembly (PCA) low-flux version of a PWR PV mockup (Tables 3, 4 and 5 and Figures 3 through 5) and the continuation of work associated with the VENUS (Fa83,Fa83a), NESDIP (Au82,Au82a,Au83), and DOMPAC (Al83) mockups (Table 1), in Belgium, the UK, and France, respectively (Fe84,Go84,Mc84, Mc84h,Ru84).

- The successful completion of the preliminary analyses for the Oak Ridge Research (ORR) Reactor simulated surveillance capsule (SSC), simulated pressure vessel capsule (SPVC) and simulated void box capsule (SVBC) LWR power plant physics-dosimetry-metallurgy experiments. Associated with this was the successful implementation and preliminary documentation of the results of an international physics-dosimetry-metallurgy "Poolside Facility (PSF) Blind Test," see Figures 7 and 8, Tables 6 through 8, and Table 10 (Gu84b,Gu84d,Li81,Mc84,St84,St84b,St84c).

- The completion of required studies associated with the evaluation and reevaluation of exposure units and values for existing and new metallurgical data bases (NRC, MPC, EPRI, ASTM, and others), Table 9 (Si84). The initial power reactor studies have involved the reanalysis of data from 47 PWR and BWR surveillance capsule reports for the W-NTD, B&W, CE, and GE power plants. Using a consistent set of auxiliary data and dosimetry-adjusted reactor physics results, the revised fluence values for E > 1 MeV averaged 25% higher than the originally reported values. The range of fluence values (new/old) was from a low of 0.80 to a high of 2.38, see also Ref (Si82a,Mc84). The research reactor studies have continued to involve the reanalysis of data originally reported by NRL and HEDL, see Ref (Mc82a), and the physics-dosimetry analysis (Li84) of the results of new test reactor (SUNY-NSTF) experiments by MEA-ENSA and HEDL (Ha83).

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- The completion of required documentation of studies associated with the data development and testing of new trend curves for the ΔRT_{NDT} shift versus neutron exposure (fluence E > 1.0 MeV and dpa) for an NRC selected power reactor surveillance capsule data base of up to 177 points; see Ref (Gu82b,Gu82c,Gu83,Gu83a,Gu83b,Gu83c,Gu84c,Mc84,Mc84h, Ra84). The status of EPRI-supported program work related to physics-dosimetry-metallurgy data development and testing is provided in Ref (Mc82c,Od78,Od79,Od83,Pe84,Va81,Va82,Va83).

- Relative to the above, and of particular interest here, has been the establishment and application of new ΔRT_{NDT} versus fluence and dpa curves for use by NRC for PTS studies (Di82,Nr80,Nr81,Nr82) and by R. Randall of NRC in the preparation of a 1984 Revision of Regulatory Guide 1.99 (Ra83,Ra84,Re77). Separate and more recent developments are the study of trend curves, based on PSF, PWR, and BWR data, that contain terms to account for possible thermal neutron, spectral and flux level effects (Mc84,Mc84h,Gu84b).

- The completion of the preparation and presentation of papers for the Fifth ASTM-EURATOM International Symposium on Reactor Dosimetry held at Geesthacht, Federal Republic of Germany, September 24-28, 1984, and the preparation and presentation of a paper at the NRC 12th Water Reactor Safety Research (WRSR) Information Meeting in October 1984, related to the PSF Experiments and Blind Test (St84c).

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2.1 ASTM STANDARDS AND PROGRAM DOCUMENTATION

2.1.1 ASTM Standards

Figures 1 and 2 provide information on the interrelationships and current schedule for the preparation and acceptance of the set of 21 ASTM standards. Results of ASTM balloting for these standards were discussed at the June 1984, Williamsburg, VA and the January 1985, Reno, NV ASTM E10 meetings. Figures 1 and 2 will be updated next at the June 1985 Toronto, Canada meeting and will be reviewed by the ASTM E10.05 Nuclear Radiation Metrology and E10.02 Metallurgy Subcommittee members to coordinate the preparation, balloting, testing, and acceptance of the entire set of standards. Refs (As84,Mc84) provide additional information related to the scope, content, and preparation of most of these standards.

E706(0) Master Matrix Guide

Lead Authors	W. McElroy (10.05)* and P. Hedgecock (E10.02)*
Participants	Lead authors of all Practices (I), Guides (II), and Methods (III)
Status	This standard will be in place in the 1985 ASTM Annual Book of Standards as E706-84. The entire standard, scope, and discussion sections were reviewed and updated.

E706(IA) Analysis and Interpretation of Reactor Surveillance Results

Lead Authors	S. Anderson and W. McElroy (E10.05)
Status	This standard will be in placed in the 1985 ASTM Annual Book of Standards as E853-84. The entire standard was reviewed and updated.

E706(IB) Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials

Lead Authors	J. Beeston (E10.02); E. Norris, and H. Farrar (E10.05)
Status	E184-79 is on the books. E. Norris and W. McElroy updated the physics-dosimetry parts of the standard for the January 1984 San Diego meeting. A title change for the standard to "Recommended Physics-Dosimetry-Metallurgy Interface Standard for LWR, FBR, and MFR Development Programs," as well as some revisions to the text were balloted at the E10.02 and E10.05 levels. As a result of this ballot and discussions at San Diego, it was reballoted for removal; the ballot results indicated the standard should not be removed. Actions on plans for revision were taken at the Reno meeting.

*P. D. Hedgecock and W. N. McElroy are the current chairmen of the E10.02 and E10.05 Subcommittees, respectively, of the ASTM E10 Committee. The current chairman of the ASTM E10 Committee is J. Perrin.

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E706(IC) Surveillance Test Results Extrapolation

Lead Authors: G. Guthrie and W. McElroy (10.05); S. Byrne (10.02)
Status: This standard will be in place in the 1985 ASTM Annual Book of Standards as E560-84. The entire contents of the standard were revised, reviewed, and updated.

E706(ID) Displaced Atom (dpa) Exposure Unit

Lead Authors: D. Doran, E. Lippincott, and W. McElroy (E10.05)
Status: This standard will appear in the 1985 ASTM Annual Book of Standards as E693-79. The need exists to update the basic nuclear data, i.e., using ENDF/B-V data and comparing the results with those obtained using ENDF/B-IV data. More complete and detailed information on the testing and application of the dpa exposure unit is provided in a Research Information Letter (RIL) on "An Improved Damage Exposure Unit, dpa, for LWR Pressure Vessel and Support Structure Surveillance," which was prepared for NRC in August 1982 (Mc82a). An ASTM news release on the results of an MPC ad hoc task group meeting on the use of dpa as an exposure unit for PV surveillance stated: "Task group members have concluded that both fluence (E > 1.0 MeV) and dpa can and should be used for the foreseeable future, until such time as the fluence (E > 1.0 MeV) is totally outmoded and no longer necessary because of appropriate standards for dpa."

E706(IE) Damage Correlation for Reactor Vessel Surveillance

Lead Authors: G. Guthrie and W. McElroy (E10.05); P. Hedgecock (E10.02)
Status: A draft of this standard has been prepared and requires further revision, which is dependent on the analysis of physics-dosimetry-metallurgy results from test and power reactor benchmarking studies in progress.

E706(IF) Surveillance Tests for Nuclear Reactor Vessels

Lead Authors: T. Mager (E10.02) and S. Anderson (E10.05)
Status: This standard will appear in the 1985 ASTM Annual Book of Standards as E185-82. An update on physics-dosimetry-metallurgy is needed in 1985-1986, particularly to address required changes associated with the use by the nuclear industry of new low-leakage core fuel management schemes. The reader is referred to the ASTM Standard E853-84 for information on needed changes in this key ASTM standard, which is used for establishing a physics-dosimetry-metallurgy surveillance program for each operating LWR nuclear power plant.

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E706(IG) Determining Radiation Exposure for Nuclear Reactor Support Structures

Lead Authors: W. Hopkins (E10.05) and P. Hedgecock (E10.02)
Status: A draft of the standard was distributed for discussion at the San Diego meeting. Appropriate revisions were made, and the standard was successfully balloted at the E10.05, E10.2, E10, and Society levels. It will be in place in the 1985 ASTM Annual Book of Standards as E1035-84.

E706(IH) Supplemental Test Methods for Reactor Vessel Surveillance

Lead Authors: R. Hawthorne (E10.02) and E. Norris (E10.05)
Status: This standard will appear in the 1985 ASTM Annual Book of Standards as E636-83.

E706(II) Analysis and Interpretation of Physics Dosimetry Results for Test Reactors

Lead Authors: F. Kam, F. Stallmann and M. Williams (E10.05)
Status: This standard will appear in the 1985 ASTM Annual Book of Standards as E1006-84.

E706(IIA) Application of Spectrum Adjustment Methods

Lead Authors: F. Stallmann (E10.05)
Status: This standard will appear in the 1985 ASTM Annual Book of Standards as E944-83.

E706(IIB) Application of ENDF/A Cross-Section and Uncertainty File

Lead Authors: E. Lippincott and W. McElroy (E10.05)
Status: This standard was successfully balloted, with appropriate editorial changes, at the E10 and Society levels. It will appear in the 1985 ASTM Annual Book of Standards as E1018-84. It is anticipated that the first version of the ENDF/A file will be issued in 1986.

A paper on the ENDF/A file and ASTM Standard was presented for the Fourth ASTM-EURATOM Symposium (Li82); another paper (Sc83) discusses the benefits and limitations of using adjusted (or benchmarked) cross sections in neutron spectrum unfolding; and Reference (As84) provides additional information on the scope of the E706(IIB) Standard.

E706(IIC) Sensor Set Design and Irradiation for Reactor Surveillance

Lead Authors: G. Martin and E. Lippincott (E10.05)
Status: This standard will appear in the 1985 ASTM Annual Book of Standards as E844-81.

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E706(IID) Application of Neutron Transport Methods for Reactor Vessel Surveillance

Lead Authors	L. Miller and R. Maerker (E10.05)
Status	This standard will appear in the 1985 Annual Book of Standards as E482-82.

E706(IIE) Benchmark Testing of Reactor Vessel Dosimetry

Lead Authors	E. McGarry and G. Grundl (E10.05)
Status	A first draft of this standard is to be submitted at the June 1985, Toronto, Canada ASTM meeting. The 1984 addition of the NBS Compendium of Benchmark Neutron Fields for Reactor Dosimetry was completed by J. Grundl of NBS and is to be distributed as an NBS publication (see Section 3.5 and Gr84a).

E706(IIF) Predicting Neutron Radiation Damage to Reactor Vessel Materials

Lead Authors	P. Hedgecock and S. Byrne (E10.02); G. Guthrie (E10.05)
Status	This standard will appear in the 1985 ASTM Annual Book of Standards as E900-83. This standard will be revised to provide trend curves based on power plant surveillance results; i.e., only power reactor data will be used to establish the curves that will be recommended for assessing and controlling the condition of pressure vessels for BWR and PWR nuclear power plants. The recent work of P. Randall (Ra84), which incorporates the results of recent studies by G. Guthrie (Gu84,Gu84c) and G. Odette (Od83,Od83a,Pe84), will form the basis for the revision of this standard.

E706(IIIA) Analysis of Radiometric Monitors for Reactor Vessel Surveillance

Lead Authors	L. Kellogg, F. Ruddy, W. Matsumoto and F. Hegedus (E10.05)
Status	This standard was successfully balloted at the E10 levels and it has been issued by ASTM as E1005-84 and will appear in the 1985 ASTM Annual Book of Standards. It makes reference to a series of other ASTM standards for the measurement of individual fission and non-fission reaction rates. The EURATOM Working Group on Reactor Dosimetry (EWGRD) is preparing a new ASTM standard for the measurement of reaction rates for the $^{93}Nb(n,n')^{93}Nb^{m}$ sensor.

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E706(IIIB) Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Vessel Surveillance

Lead Authors: R. Gold, F. Ruddy and J. Roberts (E10.05)
Status: This standard will appear in the 1985 ASTM Annual Book of Standards as E854-81. The increased application of SSTR, RM, HAFM, and DM sensors for in- and ex-vessel physics-dosimetry surveillance programs (in support of the determination of the effects of old and new fuel management schemes on the present and EOL condition of pressure vessels and their support structures) is discussed elsewhere (Mc82a,Mc84,Ru84,Ru84a).

E706(IIIC) Application and Analysis of Helium Accumulation Fluence Monitors (HAFM) for Reactor Vessel Surveillance

Lead Authors: H. Farrar and B. Oliver (E10.05)
Status: This standard will appear in the 1985 ASTM Annual book of Standards as E910-82.

E706(IIID) Application and Analysis of Damage Monitors (DM) for Reactor Vessel Surveillance

Lead Authors: A. Fudge, A. Fabry and G. Guthrie (E10.05)
Status: A draft outline was submitted. The first draft of this standard has yet to be prepared, and it is expected to concentrate on the initial use of sapphire and, possibly surveillance capsule steel correlation monitor materials. This and other candidate sensor materials for test and power reactor applications are discussed in References (Al82,Au82a,De82,Fa82,Ma82b,Pe82).

E706(IIIE) Application and Analysis of Temperature Monitors for Reactor Vessel Surveillance

Lead Authors: B. Seidel (E10.02) and G. Guthrie (E10.05)
Status: A revised draft of this standard has been prepared and is being reballoted at the E10.02 and E10 levels. It concentrates on the use of melt wires for PWR and BWR surveillance capsules.

Comments Related to the Preparation of E706(IIIE)

- S. Grant -- Improvement in the measurement of temperature in surveillance capsules is desirable to obtain second-order improvement in vessel steel radiation damage trend curves. If sufficient improvement is achieved, it becomes meaningful to also add the contribution of gamma heating. Second-order improvements are essential to projections

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that will be accurate in the 4 x 10^{19} or greater fast neutron range, as well as the <1 x 10^{19} range.

- J. Mason -- Microcalorimetry may be employed to determine gamma-ray energy deposition rates in reactor enviornments. Information on recent developments that might be relevant to the preparation of this standard are contained in Section 8.4 of the Minutes of the 13th LWR-PV-SDIP.

- A. Lowe -- The sensitivity of melt wires, or normal thermal monitors, to the actual temperature to which they are exposed means that for selected reactor operations, the monitors will record temperatures that are not indicative of normal operation. Therefore, any evaluation of thermal monitors must be done in coordination with a review of reactor history and related thermal responses.

2.1.2 Program Documentation

The following list of 22 planned NRC NUREG reports is provided for reference purposes. Each document will have Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) as the main title followed by individual subtitles. The NUREG report number and the anticipated issue date for each document are given; some subsequent annual updating of the loose-leaf documents will be required. More information and a brief statement on the contents of each document are provided elsewhere (Mc84,Mc841).

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PROGRAM DOCUMENTATION:

NRC Report No.	Vol No.	Lab Report No.	LWR-PV-SDIP Program No.*	Issue Date	Editors
NUREG/CR-1861 (PCA Physics-Dosimetry)		HEDL-TME 80-87	NUREG 1-1	July 1981	WN McElroy
NUREG/CR-3295 (PSF Metallurgy)	Vol 1	MEA-2017, Vol 1	NUREG 13-1	April 1984	JR Hawthorne
	Vol 2	MEA-2017, Vol 2	NUREG 13-2	April 1984	JR Hawthorne
NUREG/CR-3318** (PCA Physics-Dosimetry)	--	HEDL-TME 85-2	NUREG 1-2	September 1984 (Revision 9/85)	WN McElroy
NUREG/CR-3319** (Power Reactor Physics-Dosimetry)	--	HEDL-TME 85-3	NUREG 4	March 1985	WN McElroy
NUREG/CR-3320	Vol 1**	HEDL-TME 85-4	NUREG 3	September 1985	WN McElroy
(PSF SSC/SPVC	Vol 2**	HEDL-TME 85-5	NUREG 2	January 1986	WN McElroy
Experiments &	Vol 3**	HEDL-TME 86-X	NUREG 5	April 1986	WN McElroy
Blind Test)	Vol 4**	HEDL-TME 86-X	NUREG 6-1	July 1986	WN McElroy
(PSF SVBC Experiments)	Vol 5	EPRI/FCC/W-NTD	NUREG 6-4	June 1985	JS Perrin TU Marston
NUREG/CR-3320 (PSF SSC/SPVC Experiments & Blind Test)	Vol 6	CEN/SCK-XX	NUREG 6-2	December 1984	Ph VanAsbroeck JR Hawthorne A. Fabry
NUREG/CR-3321** (SDMF Physics-Dosimetry)	--	HEDL-TME 86-XX	NUREG 7	September 1986	[WN McElroy FBK Kam JA Grundl ED McGarry]
NUREG/CR-3322** (Test Reactor Physics-Dosimetry)	--	HEDL-TME 87-XX	NUREG 8	September 1987	WN McElroy FBK Kam
NUREG/CR-3323 (VENUS Physics-Dosimetry)	Vol 1	CEN/SCK-XX	NUREG 9-1	April 1986	[A. Fabry WN McElroy ED McGarry]
	Vol 2	CEN/SCK-XX	NUREG 9-2	September 1987	
NUREG/CR-3324	Vol 1	AEEW-R 1736	NUREG 10-1	January 1984	[J. Butler
(NESDIP	Vol 2	AEEW-R XXXX	NUREG 10-2	September 1985	M. Austin
Physics-	Vol 3	AEEW-R XXXX	NUREG 10-3	September 1986	WN McElroy]
Dosimetry)	Vol 4	AEEW-R XXXX	NUREG 10-4	September 1987	
	Vol 5	AEEW-R XXXX	NUREG 10-5	September 1988	
NUREG/CR-3325 (Gundremmingen Physics-Dosimetry-Metallurgy)		HEDL-TME 87-XX	NUREG 11-1	September 1987	WM McElroy
NUREG/CR-3326** (Test Reactor Metallurgy)		HEDL-TME 88-XX	NUREG 12	September 1988	WN McElroy FBK Kam

*These program numbers are not to be used on final reports.
**Loose-leaf document.

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2.2 LWR PHYSICS-DOSIMETRY TESTING IN THE ORNL POOL CRITICAL ASSEMBLY PRESSURE VESSEL BENCHMARK FACILITY (ORNL-PCA)

The pressure vessel benchmark facility at the PCA (Ka83) has afforded investigation of the following variables: 1) plant dimensions - core edge to surveillance to vessel wall to support structures positions, 2) core power distribution, 3) reactor physics computations, 4) selection of neutron exposure units, 5) neutron spectral effects, and 6) dosimetry measurements.

Results of studies completed to date indicate that routine LWR power plant calculations of flux, fluence, and spectrum, using current S_n transport methods, can be as accurate as $\pm$15% (1σ) for a criterion of $E > 1.0$ MeV if properly modeled and subjected to benchmark neutron field validation. Otherwise, errors can be a factor of two or more (Mc81). Summary information on the status of PCA program work is provided in Sections 2.2.1 and 2.2.2.

2.2.1 Experimental Program

Analysis of available passive dosimetry data has gone forward. These passive dosimetry analyses have emphasized: 1) HEDL nuclear research emulsion (NRE) measurements in the 8/7 and 12/13 configurations; 2) HEDL solid-state track recorder (SSTR) and radiometric monitor (RM) measurements to fill in and supplement former (1979-1980) measurements in the 8/7 and 12/13 configurations; 3) HEDL active gamma spectrometry measurements with the Janus probe in the 8/7, 12/13, and 4/12 SSC configurations, as well as measurements of the perturbation effects of the probe with a miniature HEDL ionization chamber; and 4) comparison of SSTR and fission chamber fission rates.

2.2.1.1 PCA Passive Dosimetry Measurements

NRE Measurements

Results of integral mode scanning of NRE irradiated in the 1981 PCA experiments have already been reported in the Minutes of the 12th LWR-PV-SDIP Meeting. An example of the comparison of these NRE data with ORNL calculations is given in Table 3 for the 1/4-T location of the 12/13 configuration. These NRE results are consistently high by almost a factor of two. Similar inconsistancies arise at the 1/2-T and 3/4-T locations of the 12/13 configuration. A detailed step-by-step examination of the emulsion technique has failed to uncover any systematic factor or effect that could account for this discrepancy. Results from the NRE experiments in the PWR engineering mockup at VENUS imply that integral mode neutron dosimetry with NRE is on firm ground. See Section 2.4.4.1 in this regard. The 1981 NRE irradiations in the PCA are now under review to determine whether any irradiation conditions could be responsible.

This disagreement could well be one of scale, since incorporation of NRE data in least-squares analyses together with RM dosimetry and calculations produces a substantial decrease in the uncertainty of the adjusted neutron spectra. More details on these least-squares analyses can be found in Ref (Mc84i), which has been issued in loose-leaf form, see Section 2.1.2. This format will permit follow-on contributions and analyses for the PCA to be included as they arise.

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SSTR Measurements

- PCA 12/13 Configuration

A comparison of measurements made in the PCA 12/13 configuration in separate sets of experiments carried out in October and November of 1981 is given in Ref (Mc84i). In general, the agreement between the two sets of data is excellent, indicating that the measurements are reproducible within the quoted experimental uncertainties.

- PCA 4/12 SSC Configuration

^{237}Np and ^{238}U fission rates were measured in the SSC, 1/4-T, 1/2-T and 3/4-T locations in the PCA 4/12 SSC configuration during January 1981. These data are summarized in (Mc84i). The relative fission rates are plotted as a function of axial location in Figure 3. All data were normalized to the midplane location. The solid line plotted for comparison is the result of Mol fission chamber traverses. The agreement of the relative SSTR fission rates with the shape of the axial distribution indicated by the fission chamber is consistent within the experimental uncertainties of the data. Fission rates as a function of radial location are provided for ^{237}Np and for ^{238}U in Ref (Mc84i). Data from the 8/7 and 12/13 configurations are also given for comparison in Ref (Mc84i) with relative uncertainties. To obtain the absolute uncertainties from relative uncertainties, the 4.1% uncertainty in the absolute power normalization must be combined in quadrature with the tabulated values. The absolute uncertainties in these data are generally $\sim$5% (1σ).

- General Data Trends

The data given in Ref (Mc84i) show that the slopes of ^{238}U and ^{237}Np fission rates (attenuation) in the PV block appear to be independent of configuration. As an additional check on the consistency of the SSTR fission rates, fission rate ratios were taken for equivalent locations in the different configurations. For the PV block, these ratios are all independent of location. Further, the relative fission rate data indicate that all the PCA SSTR fission rate measurements are selfconsistent on a relative basis and that the measurements are reproducible within the stated experimental uncertainties on an absolute basis.

- Comparison of SSTR and Fission Chamber PCA Fission Rates

At present, pending the result of benchmark irradiations of the SSTR fissionable deposits, the SSTR results are reported as absolute fission rates. The fission chamber results, on the other hand, have been benchmark referenced, and the fission chamber results are reported as fission equivalent fluxes. In order to make direct comparisons between the SSTR and the National Bureau of Standards (NBS) fission chamber results, the fission chamber data were converted into the corresponding reaction rates.

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These data, reported in Ref (Mc84i), indicate that the NBS fission chamber results are consistently about 10% higher than the SSTR results. The mean of all of the ^{237}Np ratios is 0.906 ± 0.026, the mean of the ^{238}U ratios is 0.890 ± 0.023, and the overall mean of all 22 ratios is 0.897 ± 0.025. A possible explanation for the 10% bias is the fact that the void introduced by the NBS fission chamber creates a perturbation. Measurements have been conducted at PCA to compare SSTRs, RMs, and the NBS fission chamber directly. The results of these comparisons are presently being analyzed.

Direct comparisons between the SSTRs and CEN/SCK chamber can be made only at two locations in the 12/13 configuration. No detectable bias exists between the SSTR and CEN/SCK chamber measurements at these two locations, which are both external to the PV block.

- RM Measurements

The recommended non-fission sensor [^{103}Rh(n,n'),^{115}In(n,n'),^{58}Ni(n,p), and ^{27}Al(n,p)] PCA integral reaction rates for the different configurations are given in Ref (Mc84i). A number of HEDL RM sensors were exposed at the PCA in 1981 in selected positions for the 12/13 and other configurations to complete the matrix of available RM data from the PCA experiments. The final RM, together with the NRE and SSTR results have been documented in Ref (Mc84i).

2.2.1.2 Gamma-Ray Spectrometry

To meet the needs of the LWR-PV-SDIP program, gamma-ray spectrometry has been conducted in the three low-power mockups, namely the Pool Critical Assembly (PCA) in Oak Ridge National Laboratory (ORNL) (USA), VENUS in CEN/SCK (Belgium), and NESDIP in Atomic Energy Establishment Winfrith (AEEW) (UK). A significant outgrowth of these collaborative efforts was the recognition and subsequent quantification of the perturbation factor (PF) created by the Janus probe. It was conjectured that the PF arises from the void or semi-voided regions introduced by the Janus probe into the gamma-ray intensity gradient that exists in the PV block. Initial analysis of the 1981 work performed in the 4/12 SSC configuration at the PCA had already confirmed the existence of such PF. Since the significance of this PF was clearly established, recent follow-on PF measurements at NESDIP are used here to correct the PCA measurements.

- Gamma-Ray Spectra and Dose Rates in the PCA

The PFs provided in Ref (Mc84i) have been used to correct both gamma-ray spectra and dose rates measured with the Janus probe in LWR-PV configurations studied in the PCA. However, from a rigorous viewpoint, the PF considered here are dose PF. Consequently, use of dose PF for spectral adjustments must obviously be justified. Such a justification can be made by examining spectral ratios obtained from Janus probe spectral measurements conducted at different locations of the same LWR-PV configuration. On this basis, it can be shown that the dose PF can be used for spectral adjustment without compromising experimental accuracy. This justification cannot be fully delineated here because of space restrictions, but can be found in Ref (Mc84i).

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Figures 4 and 5 compare Janus probe spectral measurements with calculations for the 1/4-T and 3/4-T locations of the 12/13 configuration, respectively. All measured gamma-ray spectra have been corrected for background as well as for the perturbation introduced by the Janus probe. Calculations for the 12/13 configuration have been performed by ORNL and CEN/SCK. These spectral comparisons are absolute and possess conventional units, i.e., gamma-rays (cm^2-MeV-s), at 1 watt of PCA power.

For the 12/13 configuration, the ORNL calculations are roughly a factor of two lower than experimental gamma-ray spectra, whereas CEN/SCK calculations occupy an intermediate position. Comparisons between theory and experiments for the 4/12 SSC configuration exhibit the same general trends. It is surprising to see that comparisons between theory and experiment generally improve with increasing penetration into the PV. However, calculations generally decrease more rapidly than experimental results with increasing gamma-ray energy.

Infinite medium dose rates $\dot{D}_{IM}$ observed with the Janus probe in the 1981 PCA experiments are enumerated in Table 4. These results have been corrected for the Janus probe field perturbation, which varies with both configuration and location. These dose rates can be taken as infinite medium dose rates in steel. It has already been shown that the difference between infinite medium dose rates for silicon and iron is negligible.

Table 5 presents a comparison of experimental and calculated gamma-ray dose rates for the 4/12 SSC configuration. In addition to the D_{IM} results from the 1981 Janus probe experiments, this table presents results obtained by the CEN/SCK group, who performed both thermoluminescent dosimeter (TLD) measurements and calculations.

Some work has continued on extending the Janus probe response matrix to higher energy. Measurements have been completed with the gamma-rays from $^{12}C^*$ ($\sim$4.4 MeV) and ^{16}O ($\sim$6.1 MeV). Analysis of these data are still underway with the goal of providing PCA experimental gamma-ray spectra up to roughly 6 MeV.

Using the results from all four locations of the 4/12 SSC configuration given in Table 5, one finds a $\dot{D}_{IM}$/TLD average ratio of 0.92. Consequently, the Si(Li) and TLD methods agree within experimental uncertainty. Comparison of these experimental results with calculations does not show consistent agreement. The extremely low calculational result at the VB location might be due to inadequate modeling of the actual geometric configuration used in the PCA. More details on these comparisons can be found in Ref (Mc84i).

2.2.2 Neutron and Gamma Calculations

2.2.2.1 PCA Neutron Calculations

Neutron transport calculations for the PCA 4/12 and the PCA 4/12 SSC configurations have been completed in support of the PSF metallurgical irradiation experiment. All neutronics calculations are performed with the DOT

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(Rh79) computer program and the VITAMIN-C (Ro82) cross-section library. The ORNL methodology utilizes a flux density synthesis technique described by Maerker and Williams (Ma82e). The purpose of these calculations is to verify that the calculations can predict the perturbation effect resulting from the insertion of a surveillance capsule. The perturbation effect is defined here to be the ratio of the ^{237}Np reaction rate with the SSC to the ^{237}Np reaction rate without the SSC. The calculations predict well the axial shape and the perturbation effect for the ^{237}Np reaction. Only relative measurements are available so that absolute comparisons cannot be made.

A paper by Lippincott (Li84a) on the evaluation of neutron flux in the PCA was presented at the Fifth ASTM-EURATOM Symposium. More detailed information on this work, and similar work by Stallman and Thomas, is provided in Ref (Mc84i).

2.2.2.2 PCA Gamma Calculations

Significant discrepancies exist in the gamma calculations between ORNL and CEN/SCK. The source of these discrepancies has been identified to be in the cross-section input. The CEN/SCK set included the gamma contributions from fission product gammas while the ORNL set did not. However, significant discrepancies between calculations and measurements still exist as discussed in Section 2.2.1.2 and shown in Figures 4 and 5.

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2.3 LWR STEEL PHYSICS-DOSIMETRY-METALLURGY TESTING

Higher flux/fluence physics-dosimetry-metallurgy benchmark fields have afforded study of the following variables: 1) steel chemical composition and microstructure, 2) steel irradiation temperature, 3) reactor operating history, 4) reactor physics computations, 5) selection of neutron exposure units, 6) dosimetry measurements, and 7) neutron spectral and dose rate effects.

A number of metallurgical programs and studies have been established to determine the fracture toughness and Charpy properties of irradiated materials as a function of chemistry, microstructure, and irradiation conditions. The ORR-PSF (Ka83) multilaboratory physics-dosimetry-metallurgy program is expected to provide key irradiation effects data, under well-controlled conditions, to help in: 1) the verification and calibration of exposure units and values and 2) the analysis and correlation of property change data obtained from this and other program work. Summary information on the status of the ORR-PSF and other program work is provided in Sections 2.3.1 and 2.3.2.

2.3.1 Experimental Dosimetry Program

For neutron dosimetry in these higher flux/fluence benchmark fields, just as for commercial LWR power plants, it is extremely advantageous to use time-integrating in addition to RM dosimeters; such as very long half-life RM, SSTR, helium accumulation fluence monitors (HAFM), and damage monitors (DM). This advantage is underscored by recent PSF calculations that show as much as 40% cycle-to-cycle variation in the saturated activities of RM dosimeters.

In addition to time-integration, another advantage in using SSTR rather than RM fission monitors is because the SSTRs make use of ^{235}U, ^{238}U, ^{237}Np, and ^{239}Pu deposits on a nickel backing disk that are of such small mass that the dosimeters can be handled as if they were just an unirradiated nickel foil; i.e., the as-received commercially available nickel backing must first be cleaned to remove the naturally occurring surface contamination of ^{238}U (with $\sim$0.7% ^{235}U content). For SSTR monitors prepared for high fluence (10^{18} to 10^{20} n/cm^2) applications, therefore, the actual amount of ^{235}U, ^{238}U, ^{237}Np, or ^{239}Pu that is electrochemically deposited on the nickel backing is less than the initial natural uranium surface contamination.

Testing and confirmation of the accuracy of RM, SSTR, HAFM, and DM sensors for LWR surveillance programs is being accomplished in PSF and SDMF physics-dosimetry-metallurgy experiments. Application of SSTR, HAFM, and DM for neutron dosimetry in higher flux/fluence LWR-PV environments entails verification and/or extension of the overall existing experimental techniques. For instance, the need for automated SSTR track scanning systems of high quantitative accuracy has been recognized for some time. The availability of such systems is an overriding factor in cost-effective SSTR applications at high flux/fluence in support of the PSF, SDMF, and PWR and BWR benchmark tests, as well as commercial applications, see Figure 6.

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2.3.1.1 PSF Physics-Dosimetry-Metallurgy Experiments

The 2-year physics-dosimetry-metallurgy irradiation experiment in the ORR-PSF was completed June 22, 1982. The SPVC and the SVBC were disassembled, and the dosimetry sensors and metallurgical specimens were shipped to the appropriate participants. The preliminary physics-dosimetry-metallurgy, irradiation and temperature distribution data, and the reactor power time-history data for these LWR-PV steel simulation experiments have been documented; the results were made available to participants of the PSF Blind Test Workshop held in April 1984, at Richland, WA.

The preliminary HEDL and ORNL physics-dosimetry results have been compared with those obtained by other participants (Belgium, UK, Germany, and US) and no significant discrepancies exist. Final exposure parameter values (fluence-total, thermal, E > 1.0 MeV; and dpa maps) for SSC-1, SSC-2, SPVC, and SVBC must still have the concurrence of all program participants doing physics-dosimetry analysis.

The final MEA metallurgy results and analysis for SSC and SPVC have been documented and reported in NUREG/CR-3295, Volumes 1 and 2, by R. Hawthorne et al, (see Section 2.1.2) (Ha84,Ha84a). Copies of these reports were made available to the participants of the PSF Blind Test Workshop in April 1984.

The program documentation plan and schedule for all of the PSF Experiment and PSF Blind Test NUREG reports (six volumes) are given in Section 2.1.2.

2.3.1.2 PSF Experimental and Blind Test

The ORR-PSF experiment was specifically designed to simulate the surveillance capsule-PV configuration in power reactors and to test the validity of procedures that determine the radiation damage in the vessel from test results of surveillance capsules. Emphasis was on radiation embrittlement of reactor vessel steels and damage correlation in order to test current embrittlement prediction methodologies. For this purpose, a PSF metallurgical Blind Test was initiated (Mc83d). Experimental results were withheld from the participants; only the type of information normally contained in surveillance reports was given. The goal was to predict from this limited information the metallurgical test results in the PV wall capsule. Of particular interest was the question: What effects, if any, the differences in fluence rate and fluence spectrum in the surveillance capsule and in the PV wall may have on embrittlement predictions?

To serve as a benchmark, a very careful characterization of the ORR-PSF experiment was necessary, both in terms of neutron fluence spectra and of metallurgical test results. Statistically determined uncertainties must be given in terms of variances and covariances to make the comparisons between predictions and experimental results meaningful. A more detailed description of the PSF experiment characterization program and the preliminary results of the PSF Blind Test are given in Ref (St84c). Here, only preliminary results of the Blind Test and its implications will be highlighted.

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- Description of the PSF Experiment

The ORR-PSF (Figure 7) consists of the ORR reactor core and the ex-core components that are used to mock up PV surveillance configurations for LWRs. The ex-core components are the thermal shield (TS), the SSC, the SPVC, and the SVBC. The aluminum window is part of the ORR PV that separates the core from the ex-core components. The PSF metallurgical Blind Test is concerned with the SSC and the SPVC. Five metallurgical specimen assemblies were prepared for the irradiation experiment. Each assembly (Figure 8) contains the same mix of plate, forging, and weld material specimens (Ha84,Ha84a). Dosimeters are distributed throughout each assembly to monitor the neutron exposure received by the specimens. Two capsules were fabricated for irradiation in sequence at the simulated surveillance locations (SSC-1 and SSC-2) to fluences (E > 1.0 MeV) of $\sim 2 \times 10^{19}$ and $\sim 4 \times 10^{19}$ neutrons/cm^2, respectively. Each SSC contained one of the metallurgical specimen assemblies. The SPVC contained the other three assemblies, which were positioned at locations corresponding to the inner surface (0-T), the quarter thickness (1/4 T), and the half thickness (1/2 T) of a PV. The fluences for SSC-1 and SSC-2 are approximately equal to the 1/4-T and 0-T positions, respectively. The total irradiation times for SSC-1 and SSC-2 are approximately 46 days and 92 days, respectively, while the irradiation time for the SPVC is ~600 days. The temperature of the specimens was tightly controlled to 288°C $\pm$ 7°C during the irradiation (Mi81).

- Neutron Transport Calculations

Flux, fluence, and reaction rate calculations were performed for each of the three exposures (two surveillance capsules and a PV capsule) performed during the two-year metallurgical Blind Test experiment at the ORR-PSF. Motivation for these calculations was prompted by differences of up to 25% between dosimetry measurements performed in the earlier startup scoping experiment and the two-year experiment.

Following the same simplified calculational methods used in a re-analysis of the startup experiment, fission source distributions were obtained from the three-dimensional (3D) diffusion theory for most of the 52 cycles active during the course of the complete experiment, combined in small groups, and the resultant ex-core group fluxes calculated by the two-dimensional (2D) discrete ordinate transport theory. More details can be found in Ref (Ma84a) and (Ma84b).

Comparisons of the ORNL-calculated dosimeter end-of-irradiation activities with HEDL measurements indicate agreement generally within 15% for the first surveillance capsule, 5% for the second, and 10% for three locations in the PV capsule, which are as good as, if not somewhat better than, comparisons in the startup experiment. The calculations thus validate the trend of the measurements in both the startup and the two-year experiments and confirm the presence of a significant cycle-to-cycle variation in the core leakage.

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The tape containing the unadjusted spectral fluences for each of the three exposures that can be used in the metallurgical analysis is thus considered to be accurate to within about 10%.

- Dosimetry and Adjustment Procedures

A 10% accuracy, as quoted in the preceeding section, for the damage parameter values of the metallurgical specimens is quite sufficient for most metallurgical damage correlation studies. However, since the ORR-PSF experiment is intended to be a benchmark, higher accuracies and a more thorough study of the uncertainties is required. Thus, comprehensive statistical analyses with the use of adjustment procedures are being made by program participants to obtain complete three-dimensional fluence-spectrum maps. These maps will include not only the damage parameter values $\phi t > 1.0$ MeV, $\phi t > 0.1$ MeV (ϕt = fluence), and dpa, but also reaction rate values for all major threshold reactions. These reaction rate values are included to test dosimetry measurements from a variety of laboratories and some experimental dosimetry (e.g., damage monitors) which were not used in the adjustment procedure.

Details of the dosimetry sensors in the ORR-PSF experiment are given in Ref (Li81). Summary information on the analysis of physics-dosimetry for the ORR-PSF experiment by Blind Test participants is given in Ref (Gu84d). The measurements, which are available to date, can be found in the two references. Both the LSL-M2 and FERRET adjustment procedures (St84a,Sc79) have been used for the evaluation. The ORNL results are discussed in Ref (St84) and the HEDL and combined HEDL/ORNL results have yet to be documented; in general, exposure parameter value uncertainties are in the 5 to 15% (1σ) range.

- Statistical Evaluation of the Metallurgical Tests

The need for rigorous statistical evaluation of the experimental results is not restricted to dosimetry and neutron fluence determination. The high standard for accuracy and reliable determination of uncertainties applies also to the metallurgical test results. The primary source for the Blind Test comparisons is the Charpy test results. The raw Charpy data (Ha84) are fitted to continuous curves, impact energy vs test temperature, in order to determine the shift of nil-ductility temperature (ΔNDT) and upper-shelf energy (ΔUSE) with increasing damage fluence. Hand-drawn fits are used in Refs (Ha84) and (Ha84a) but without assigned uncertainties. The HEDL evaluation makes use of hand-drawn and least-squares fit (Gu84b). In the ORNL evaluation (St84b,St84c), a computer fit CV81* was used. This fitting procedure uses separate curves for ΔNDT and ΔUSE and is, therefore, more flexible than hyperbolic tangent or error function approximations, which are commonly used for computer fits. (These fits were used by some Blind Test participants.) There are no obvious differences between the CV81 evaluations and

*CV81 is a linear least-squares procedure, although linear combinations of non-linear functions can be used.

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the hand-drawn fits in Ref (Ha84), as can be seen in Table 6, but the statistical computer fit allows the calculation of uncertainties. Table 7 lists the different materials used in the metallurgical irradiation and their chemical composition. A summary of the CV81 results is given in Table 8. Details of the procedure are given in Refs (St84b,St84c).

- Discussion of the Blind Test Results

The participants of the Blind Test received the following information (Mc83d):*

1) Calculated 102-group flux-spectrum, exposure rate parameters, and dosimeter reaction rates for the SSC-1, SSC-2, and SPVC 0-T PV surface, 1/4-T wall, 1/2-T wall, 3/4-T wall, and SVBC positions.

2) The SSC-1 measured in-situ dosimetry as-built information, exposure time history, and post-irradiation sensor results.

3) The SSC-1, SSC-2, and SPVC 0-T, 1/4-T, and 1/2-T measured in-situ Co-Al alloy bare and gadolinium-covered sensor results used to determine two additional low-energy group flux-spectral values for the thermal and thermal to 9.8×10^{-2} MeV energy ranges.

4) The SSC-1, SSC-2, and SPVC measured metallurgical specimen exposure time histories.

5) The SSC-1, SSC-2, and SPVC measured metallurgical specimen temperature time histories.

6) The SSC-1, SSC-2, and SPVC as-built metallurgical specimen dimensional and placement information.

7) The SSC-1 metallurgical specimen heat treatment, chemistry, and measured unirradiated and irradiated properties for different steels.

Participants were asked to predict both the damage parameter values and the metallurgical test results in the SSC-2 and SPVC capsules.

To determine damage parameter values, most participants used the calculated fluences normalized with measurements at the SSC-1. Adjustment procedures and cosine-exponential fits were also used by some participants. Uncertainties were quoted by some participants and were all on the optimistic side. None of the quoted figures for damage parameter values differed by more than 30% in either direction from the ORNL evaluation, and 65% of the values were

*This information will be made available to those who have an interest in making their own analysis, predictions, and contributions to the PSF experiment studies.

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within ±10 of ORNL. Differences in the damage parameter determinations had very little impact on the determination of radiation damage. That is, some participants who predicted low parameter values quoted high embrittlement values and vice versa.

The prediction of materials property changes, primarily NDT and USE for Charpy vs fluence, were all based on one of the two formulas

$$\Delta M = C\ (\phi t)^{a} \quad , \text{ or} \tag{1}$$

$$\Delta M = C\ (\phi t)^{[a - b \cdot \log(\phi t)]} \tag{2}$$

with ΔM materials change and ϕt fluence > 1.0 MeV (or some other damage parameter such as dpa). C is a "chemistry factor" that is either determined explicitly from the chemical composition and a data base or used as a scale factor based on the SSC-1 results. Formula (1) with a = 0.5 is used in NRC Reg. Guide 1.99 (Re77). Other (usually smaller) values of a are used by some participants as obtained from their data bases. The "Guthrie formula" (Gu84) replaces the straight line in a log-log plot by a parabola, taking into account that damage "saturates" faster than a single exponent would indicate. The parameters a and b were either the ones originally obtained by Guthrie or modifications obtained from their own data bases.

The MEA and ORNL 41J-ΔRT_{NDT} results were compared with Blind Test predictions and summarized in Table 6. Lowest and highest predictions are strikingly close to each other and mostly symmetrically distributed relative to the experimental values. The largest deviation, between measurements and prediction, is for the weld (Code R) in the SPVC capsules. The high nickel content places this material outside the data bases from which the prediction formulas were obtained. Aside from this material, no consistent biases nor major deviations between predictions and experimental values were found. Also, none of the prediction formulas was consistently superior. The explanation for this outcome is discussed in Ref (St84c).

To explain the differences in trend curves for different materials quantitatively, much more needs to be known about the physical processes that lead to radiation embrittlement. This can be obtained only with more detailed studies of the microstructure. The current information from this experiment and other metallurgical data bases is purely phenomenological. Any attempt to fit such data to some simple model results in fairly large data scatter, which is not likely to improve by adding more parameters to the model. In the Blind Test predictions, a 1σ uncertainty of approximately ±20°C for ΔNDT was observed, and this uncertainty was about the same for different prediction formulas. The uncertainty is likely to be somewhat larger for predictions based on surveillance specimens in commercial power reactors. More work is needed to establish reliable safety margins in these cases (Gu84c).

The reader is referred to Ref (St84c) for more information on the comparison of the metallurgical results from the ORR-PSF experiment with the predictions of Reg. Guide 1.99, Revision 1, (Re77), which is based on formula (1) and the

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proposed Revision 2 (Ra84), which is based on the Guthrie formula (Eq. 2) (Gu84). Additional information on the effects of irradiation temperature, fluence rate, neutron spectrum, and thermal neutrons are provided in Ref (Gu84b, Mc84h, and/or St84c).

- Application to PV Surveillance and Regulatory Procedures

These preliminary PSF metallurgical Blind Test comparisons show that the various embrittlement prediction formulas are adequate as rough approximations, but that none captures the complex and not-well-understood correlation between radiation embrittlement and fluence, fluence rate, neutron spectrum, chemistry, heat treatment and, perhaps, other factors. None of the current prediction formulas appears clearly superior, but the Guthrie formula captures somewhat better the saturation effect at higher fluences, since the quadratic term adds more flexibility. Improvements are possible in the following areas:

1) Realistic uncertainties need to be established in conjunction with prediction formulas. The Blind Test comparison gives some clues for the size of uncertainties (Table 6). Larger data bases and statistical evaluations, which are more specifically directed towards uncertainties, are needed for more definite results, see Ref (Gu84c).

2) Prediction formulas, which are derived from a data base, may not be valid for materials whose composition is outside the range of the data base. Substantially higher safety margins must be applied to such materials and/or test irradiations must be performed to establish trend curves for the particular material.

- Recommendations for Future Research

The PSF experiment has essentially confirmed the broad trends for radiation damage which had been determined from existing data collections. Underneath these general trends lies, however, a bewildering variety of detailed structures that 1) reflect the complexity of the damage mechanisms and 2) contributed to substantial uncertainties in predicting radiation damage. Any improvements in prediction will come only from a better understanding of the effects of radiation to the microstructure of the material and from a precise quantification of these effects. Future research planning should consider experiments that are directed to a better understanding of the embrittlement mechanisms and the combined (and separate) effects of the involved variables.

2.3.1.3 ORR-SDMF

In addition to verification of surveillance capsule perturbation effects, the ORR-SDMF (Ka83) tests provide benchmark referencing of the primary neutron sensors used for irradiation surveillance of pressure vessels and their support structures. The SDMF tests are conducted in the high-flux environment of the PSF adjacent to the ORR. These tests and the SDMF Facility are an outgrowth of the LWR-PV-SDIP. They are a result of the need

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1) to benchmark calculations and QA dosimetry sensor materials in flux environments more intense than are available in pure standard fields and 2) to acquire data to validate and substantiate methods, procedures, and data recommended for use in the ASTM standards.

Results of the Westinghouse-Combustion Engineering Surveillance Capsule Perturbation Experiment (2nd SDMF test) were reported in the 1982 Annual Report (Mc82a). Experimental results from the B&W Surveillance Capsule Perturbation Experiment (3rd SDMF Test) are not yet available. HEDL and other program participants are still in the process of completing their RM sensor and physics-dosimetry analysis for the 3rd test.

Work continued in FY84 on the physics-dosimetry analysis for the November 1983 4th SDMF test; a nominal 18-day irradiation of selected RM (HEDL, NBS, KFA), SSTR (HEDL), HAFM (RI) and DM (Harwell-RR&A) sensors in the 4/12 configuration with an SSC attached to the back of the TS. Ref (Mc84) provides additional information on the RM, SSTR, HAFM, and/or DM sensors selected for irradiation in the 4th test.

Preliminary results of the UK analysis of the sapphire DMs for the PSF and 4th SDMF test were discussed by A. Fudge in a poster session and a workshop at the 5th ASTM-EURATOM Symposium. These PSF and 4th SDMF-DM monitor results correlated well with other test reactor physics-dosimetry results when plotted versus measured $^{93}Nb(n,n')$ reactions.

A special "tungsten photo-fraction gauge experiment" was placed at the back of the 4th SDMF void box to obtain some information about photofission corrections to RM and SSTR fission reaction rates in a cavity-like environment. Ref (Ve80) provides more information on photofission corrections.

NBS-certified neutron fluence standards were sent to HEDL and KFA for RM sensor counting with dosimeters from the 4th SDMF irradiation. The nuclear reactions involved are: $^{238}U(dep)(n,f)FP(Ba\text{-}La)$, $^{58}Ni(n,p)^{58}Co$, and $^{54}Fe(n,p)^{54}Mn$.

2.3.1.4 BSR-HSST

The Bulk Shielding Reactor (BSR) metallurgical results of the 61W to 67W series for the Heavy-Section Steel Technology (HSST) Program have been reported in Refs (St82d,St82e) by ORNL. The original computer program for the statistical analysis has been modified and generalized to include nonlinear fitting. Additional information is provided in Refs (Fa80a,Ka82,Ka82a,Ka82b).

2.3.1.5 SUNY-NSTF

A joint MEA-ENSA-HEDL metallurgical irradiation study is underway with metallurgical specimens being irradiated at the State University of NY (SUNY) Nuclear Science and Technology Facilities (NSTF) at Buffalo, NY.

The purpose of the experiments is to determine the effect of variations of chemical composition on the irradiation embrittlement sensitivity of alloys

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having a composition typical of reactor PV steels. To determine the effects of the variations of the individual elements, a base composition has been selected and extra concentrations of particular elements have been added, one, two or three elements at a time.

MEA is responsible for melts, experiment design, construction, irradiation, and Charpy/tensile tests. HEDL is responsible for small specimen compression and hardness tests, fractography and computer analysis data/interactions and the physics dosimetry characterization program.

To date, 7 main melts have been prepared and split into 4 chemically different ingots in each main melt for a total of 28 separate ingot compositions. Specimens from 16 of the 28 ingots have been irradiated and tested (Charpy), and specimens from an additional 8 ingots have been irradiated but are at present untested. The results of the initial tests are available in Ref (Ha83). The results identify phosphorous as a detrimental element. A phosphorous saturation phenomenon was observed.

Analysis of the Buffalo dosimetry test has been completed and fluxes have been derived using the dosimetry results and the HEDL-ORNL calculated flux spectrum. A paper on this work was presented at the 5th ASTM-EURATOM Symposium (Li84). Good agreement between the dosimetry results and the calculation was obtained. Further details on the calculated and measured results will be reported in the next LWR-PV-SDIP semiannual progress report, including axial profiles and estimated gradients through the specimen assembly.

Two dosimetry capsules from irradiation UBR-51 have been analysed. The results will be correlated with the dosimetry test results and calculated flux spectrum and reported in a semiannual progress report for FY85.

Three additional capsules have been supplied by HEDL to MEA for inclusion in a metallurgy experiment. Schedule for recovery and analysis of these capsules is not known at this time. It is also expected that HEDL dosimetry results for the MEA irradiation will be correlated with independent dosimetry measurements made by J. W. Rogers (EG&G, Idaho).

2.3.1.6 LWR Power and Test Reactor Advanced SSTR Dosimetry

HEDL is in the process of completing the preliminary analysis of SSTRs (^{238}U, ^{237}Np, etc.) irradiated in the PSF Experiments to verify their applicability for use in very high fluence (10^{18} to 10^{20} n/cm^2) test and power reactor metallurgical specimen irradiation capsules.

WHC and NBS through NRC and subcontracts to W-NTD and B&W is now providing state-of-the-art advanced cavity dosimetry for commercial LWR operations (Ru84,Ru84a). This dosimetry provides improved measurement of neutron fluence for reactor specific evaluations related to PV embrittlement. SSTRs are one of the key dosimeters, and their effective economic use requires an automated system for counting tracks up to and beyond 10^7 tracks/cm^2, see

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Figure 6. Recent attempts to scan such high density SSTRs have been quite promising and indicate that accuracies of better than 5% (1σ) may be attainable at these very high track densities.

The development of advanced SSTR scanning (counting) systems for routine LWR power and test reactor dosimetry applications is particularly significant because, as stated previously, the SSTRs make use of ^{235}U, ^{238}U, ^{237}Np, and ^{239}Pu deposits that are of such small mass that the un-irradiated dosimeters can be handled as if they were just un-irradiated nickel foils.

2.3.2 Neutron Calculations

2.3.2.1 PSF Neutron Calculations

Flux, fluence, and dosimetry calculations were made of the two-year metallurgical Blind Test irradiation experiment performed at the ORR-PSF during the period from April 1980 to June 1982.

Early in the calculations, it became apparent that significant cycle-to-cycle variations (up to ~40%) could exist in the ORR core neutron leakages among the 52 cycles active in the irradiation. In order to compare dosimetry calculations with measurements, few short cuts could be employed. Nothing short of a complete analysis, taking into consideration the source distribution of each of the 52 cycles as well as their leakages, would suffice if an accurate comparison were desired. Initial comparisons of calculated results with HEDL dosimetry measurements have been performed for the experiment. It is to be observed that the variation is as much as 40%, with cycle groups 158C + 158D and 161C representing the extremes. The spectrum remains unaltered from cycle to cycle.

2.3.2.2 PV Wall Neutron Fluence Attenuation Predictions

The preliminary results of PSF comparisons lie within 10%, but reaffirm slight deficiencies in the iron cross sections first brought to light by the PCA and PSF Startup experiment comparisons (Mc81,Wi83), which show increasing disagreement the further into the PV one goes.

In the planned Revision 2 of Reg. Guide 1.99 (Ra84), the equation used for PV wall fluence attenuation by Randall is

$$\text{Fluence}(X) = \text{Fluence}(\text{Surface}) \cdot e^{-0.24X}, \qquad (3)$$

where X is the depth in the wall in inches, measured from the inside surface. This equation is based on transport calculations by Guthrie et al. (Gu82, Gu82a) for the dpa attenuation through an 8.0-inch vessel wall. These calculations did not account for the deficiencies in the iron cross sections mentioned above.

It has been recently noted by Fabry that the $^{6}Li(n,\alpha)$ spectrometry data (DeLeeuw) (Mc81) in PCA are consistent with gas proton recoil spectrometry (Rogers) (Mc81) and silicon damage measurements (DeLeeuw) (Mc81), and they indicate larger proportions of neutrons below 1.0 MeV than predicted by

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ENDF/BIV; the discrepancy is of the order of 20%, in the same direction as NRE results reported by Roberts, Gold, and Preston in Section 2.2.1.1 - NRE Measurements. This confirmed result does affect the dpa/ϕ>1 MeV transverse predictions through the reactor PV planned for use in Reg. Guide 1.99, Revision 2 (Ra84), and may adversely impinge upon eventual crack-arrest considerations in the safety analysis of ASME-III designed vessels. It is recommended, therefore, that:

1) A new simultaneous evaluation of all experimental data in PCA, the NESDIP replica, and the Mol Iron Shell Benchmarks should be performed, including the French DM results obtained during the PSF start-up program,

2) Integral measurements using NRE as well as higher threshold-energy sensors [such as ^{58}Ni(n,p) or ^{64}Zn(n,p), ^{27}Al(n,α)] should be performed in the Mol Iron Shell Benchmarks, and

3) Continuous gamma-ray spectrometry experiments be conducted in the NESDIP benchmark, Phase 3, to resolve inelastic gamma-rays produced by fast neutron interactions in iron and thereby test the inelastic neutron transport cross section of iron.

2.3.2.3 ORR-SDMF

Westinghouse plans to complete the transport calculations and analysis for the W and CE surveillance capsule perturbation experiment by July 1985.

The calculational program to determine the energy-dependent flux distribution throughout the test region for the B&W surveillance capsule perturbation experiment was started by C. Whitmarsh of B&W. The cross-section library to be used by ORNL to compute the source distribution for the B&W perturbation experiment was completed and the results were made available to B&W.

2.3.2.4 BSR-HSST

The BSR-HSST irradiation experiments have been completed and the results have been documented (Be83,St83).

2.3.2.5 SUNY-NSTF

HEDL has the lead responsibility for modeling, completing, and documenting the results for the transport calculations for the SUNY-NSTF (Buffalo, NY) MEA-ENSA-HEDL chemistry-metallurgical tests. ORNL has provided technical assistance in the use of the DOT transport code and offered suggestions as to the modeling of the core and experiment. MEA-ENSA have provided detailed information on the Buffalo irradiation rigs and their operation (i.e., materials, geometries, dimensions, tolerances, water, and air gaps changes resulting from temperature control, thermocouple lead gaps, etc.). These HEDL-ORNL calculations were completed in FY 1984. HEDL has used the FERRET code to obtain preliminary dosimetry adjusted neutron exposure parameters for this important series of metallurgical irradiations. A paper on the physics-dosimetry was prepared and presented at the 5th ASTM-EURATOM Symposium by Lippincott et al. (Li84).

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2.4 ANALYSIS AND INTERPRETATION OF POWER REACTION SURVEILLANCE AND RESEARCH REACTOR TEST RESULTS

Summary information is presented in the 1983 Annual Report (Mc84) on the results of previous LWR-PV-SDIP studies associated with physics-dosimetry-metallurgy data development and testing for power reactor surveillance and research reactor irradiation effects programs. Updated information is provided in the following Sections.

2.4.1 Surveillance Capsule Data Development and Testing

2.4.1.1 Trend Curve Data Development

As discussed in the 1983 Annual Report, and as a part of the LWR-PV-SDIP Program, statistically based data correlation studies have been made by HEDL and other program participants using existing PWR and BWR physics-dosimetry-metallurgical data in anticipation of the analysis of new fracture toughness and embrittlement data from the BSR-HSST, SUNY-NSTF, ORR-PSF and other experiments. The reader is referred to Ref (Mc84) for additional summary type information and appropriate references.

In Ref (Si84), Simons presents results of evaluation and reevaluation of exposure units and values for 47 PWR and BWR surveillance capsule reports for W, B&W, CE, and GE power plants. Using a consistent set of auxiliary data and dosimetry-adjusted reactor physics results, the revised fluence values (Table 9) for E > 1 MeV averaged 25% higher than the originally reported values. The range of fluence values (new/old) was from a low of 0.80 to a high of 2.38, see also Ref (Si82a,Mc84). These HEDL-derived FERRET-SANDII exposure parameter values have been used for the HEDL PWR and BWR trend curve data development and testing studies discussed in subsequent sections of this report.

In Ref (Ra84), Randall discusses the basis for his Revision 2 of Reg. Guide 1.99. As stated, the Guide is being updated to reflect recent studies of the physical basis for neutron radiation damage and efforts to correlate damage to chemical composition and fluence. Revision 2 contains several significant changes. Welds and base metal are treated separately. Nickel content is added as a variable and phosphorus is removed. The exponent in the fluence factor is reduced, especially at high fluences; and guidance is given for calculating attenuation of damage through the vessel wall [see Equation (3), Section 2.3.2.2].

In Refs (Gu84b) and (Mc84h), the effects of changes in different variables and use of different exposure parameter models for predicting the Charpy shift for the 30-point PSF weld, plate, and forging data base and a 30-point PWR weld data base are discussed in considerable detail.

The main comments and conclusions of Guthrie's study (which is based on the use of PSF and test reactor data) are:

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1) In surveying the previously existing data available for the alloys in the PSF experiment, it has become apparent that the fluence exponent is dependent on temperature and flux level. For A302B alloy, the PWR surveillance data fell consistently below the higher flux level LITR data and showed a lower value for the fluence exponent. The overall scatter of the existing data is such that it is not clear that Charpy tests or K_{IC} tests can be used to uncover fine details in mechanisms.

2) Because of the possible rate effect (which was predicted by G. R. Odette in his PSF Blind Test submission), the PWR suveillance trend curve laws cannot be expected to work as well in the PSF as might be expected from their stated standard deviations.

3) In applying existing Charpy shift laws to the PSF C_V data, we find that the largest observed shift occurred for the Rolls Royce A533B weld (Code R), which had a high nickel content (1.58%) - well outside the range of the data base used to develop the HEDL PWR Charpy shift equations (Gu84). A comparison of the HEDL equation applications and the Hawthorne values for ΔT_{Cv30} are shown in Table 10. The overall deviation is 31.6°F or 17.6°C (1σ). This is more than the standard deviation of the fit to the original data base, and is due to the facts that 1) the Code R specimen is outside the recommended chemistry range, and 2) the rate effect has caused the predictions for ΔT_{Cv30} to be biased low. The values in Table 10 should not be compared to blind test predictions since no use was made of SSC-1 results to guide the calculations in Table 10, and no correction has been attempted for rate effects.

4) There appears to be a rate effect in the PSF Charpy and compression data. The fluence exponent appears to increase with increased flux and appears to decrease with increased copper.

5) The similarity of the spectra at the separate irradiation positions severely limits the possible comments about damage functions.

6) No extra thermal neutron effect, beyond that already represented in the ASTM dpa cross section was identifiable in the PSF data.

The main comments and conclusions of McElroy's et al. study (which is based on the use of PSF, PWR, and BWR data) are:

1) There is a significant improvement (reduction) in the standard deviation of the fit for weld Charpy shift trend curves that include the effect of low energy thermal neutrons. For the 30-point weld data set, improvements of the amounts observed could occur at a frequency of ~4% by chance.

2) A knowledge of the actual boron content of PV steels and the use of a trend curve that makes use of an exposure parameter dose term, which includes the total production of dpa in iron and helium, could make significant improvements in lowering the standard deviation of the fit for the existing PWR surveillance capsule metallurgical weld data base.

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3) Based on the trend curve model that includes the effect of thermal neutrons, for both PWR and BWR power plants, up to about 80% of the SS clad/PV steel wall interface and surveillance capsule specimen dose term values could be attributed to helium production in PV steels, depending on the particular surveillance capsule design, Charpy specimen placement, steel boron content, and power plant operating conditions.

4) Existing PWR and BWR surveillance capsule derived embrittlement trend curves [based on the use of just fast fluence (E > 1 MeV) or dpa for the exposure term] cannot be expected to give reliable predictions of the combined fast and thermal neutron contributions to the Charpy shift at the SS clad/PV steel wall interface, 1/4-T, 1/2-T, 3/4-T, or 1-T locations. [It is noted that the PSF experiment provides physics-dosimetry-metallurgy data for predicting the Charpy shift in PV steels at deep in-wall locations, such as the 1/4-T, 1/2-T, and 3/4-T positions, where the T/F ratios are in the very low range of $\sim$0.14 to $\sim$0.53. However, even for these very low ratios, helium from both boron and steel high energy (n,α) reactions may still contribute 5 to 30% to the exposure parameter dose term value.]

5) None of the Charpy shift trend curve equations studied, Table 1 of Ref (Mc84h), except perhaps the one based on the use of an exposure parameter of fluence E > 0.1 MeV, appear to properly bound all of the six PV steel observed PSF damage gradient curves. Based on the French simulated PV-wall DOMPAC Experiment (Mc84,A183), Alberman concluded that for low temperature (<100°C) irradiations, fast fluence (E > 1 MeV) is too "optimistic" and is not, therefore, a conservative neutron exposure parameter. That, at low temperature, 95% of the measured damage (based on tungsten and graphite DM results) comes from neutrons with energy E > 0.1 MeV. This led him to conclude that the exposure parameter, fluence (E > 0.1 MeV), is perhaps "pessimistic", but has the advantage of being the lower threshold of all (displacement) damage models and thus it takes into account all neutrons that create (displacement) damage.

6) The plant specific weld data sets used in the PWR and BWR data base studies, except for one, do not support a saturation effect at high fluences above $\sim 1 \times 10^{19}$ n/cm^2 (E > 1 MeV). Consequently, the existing Reg. Guide 1.99 (Re77) upper bound (truncated) trend curve model shape (or plant specific curves) may have to be used for high fluence embrittlement predictions for PV steel welds, and perhaps forgings and plates.

7) Any significant thermal neutron contribution to PV steel embrittlement is, most probably, a result of (n,α) reactions in boron-10 rather than by neutron-induced $Fe(n,\gamma)$ recoil reactions.

8) It appears that the current ASTM E693 dpa cross section should not be used to correlate highly thermalized light or heavy water moderated power or test reactor irradiation effects data because it significantly overestimates the low energy thermal neutron dpa contribution.

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9) The PV-wall SS clad/PV steel interface surface T/F ratio for PWR and BWR power plants is expected to be in the range of 2 to 6 on the basis of surveillance capsule measurements, Westinghouse transport calculations, GE measurements, and the PSF experiment physics-dosimetry results.

10) Individual Charpy specimens (with natural boron content of ~0.4 up to perhaps 5 wt ppm) in PWR and BWR surveillance capsules will be subject to ϕ neutron exposures with T/F ratios in the range of ~0.5 to 5, depending on the surveillance capsule design, its placement and the reactor operating conditions. The T/F variation for individual Charpy specimens, therefore, could be an important parameter for the correlation of a set of Charpy specimen results and derived ΔRT_{NDT} values.

11) From this study, that of Grant and Earp (Gr84), and others discussed in Ref (Mc84h), a final conclusion is: the PSF experiment and PWR and BWR surveillance program results clearly show that comparison of the effects of radiation damage on yield strength, hardness, RT_{NDT} and USE will be needed to aid in improving and refining our knowledge of trend curves and PV wall damage gradients. Implicit in this are the current observations that the establishment of separate trend curves for welds, forgings, and plates will give increased understanding and accuracy in projections of the present and future metallurgical condition of PV steels.

2.4.2 Research Reactor Data Development and Testing

As discussed in previous annual reports and as a part of the LWR-PV-SDIP Program, statistically based (as well as other) physics-dosimetry-metallurgy data analysis and correlation studies using research reactor data are being made by ORNL, MEA, HEDL, UCSB and other program participants. The reader is referred to Sections 2.3.1 and 2.3.2 and the proceeding of the 5th ASTM-EURATOM Symposium for more information on the ORNL, MEA, HEDL, UCSB, and other studies.

2.4.3 Benchmark Referencing Program

Benchmark referencing studies on both the experimental and calculational aspects of the LWR-PV-SDIP are important program elements. The results of such studies are discussed and referenced throughout Sections 1.0 and 2.0. Section 3.0 provides additional information about FY84 accomplishments by NBS and other LWR-PV-SDIP participants related to benchmark referencing and dosimetry calibration activities for research and power reactor test and surveillance programs. More complete information is given in Ref (Mc84) and in the proceedings of the Fifth ASTM-EURATOM Symposium.

2.4.4 VENUS, NESDIP, and DOMPAC Benchmark Experiments

2.4.4.1 Mol, Belgium VENUS PWR Core-Baffel-Barrel-Thermal Shield Benchmark

Dosimetry experiments in the PWR engineering mockup at the VENUS critical facility were carried out in the first half of 1983. A detailed description

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of the VENUS facility at CEN/SCK, Mol, Belgium, can be found in Ref (Fa83, Fa83a,Mc84). This mockup was established to provide a relevant and practical reactor physics link between PCA/PSF tests and actual environments of LWR power plants. Indeed for actual power plants, the azimuthal and vertical variation of the surveillance capsule lead factors can not be ignored. These variations, together with the core boundary fuel power distribution must be treated in detail, otherwise undetected biases may be entailed in calculations. Such biases will be further exacerbated by the use of advanced fuel and core management (low leakage) schemes, where the effects of fuel burnup and plutonium build-in must be handled properly to obtain reliable reactor transport calculational results. A CEN/SCK paper on the results of the VENUS program work was presented at the Fifth ASTM-EURATOM Symposium (Fa84).

Work is in progress to complete the analysis of the measurements performed in VENUS in 1983. A second series of ^{6}Li neutron spectrometry measurements and irradiations of SSTRs were further performed in April-May 1984. Draft sections of NUREG/CR-3323, see Section 2.1.2, on the VENUS experiments have been completed and have been distributed to program participants.

Results of transport theory calculations by CEN/SCK (Fa84), ORNL (Wi84b) and Westinghouse (Fe84) were presented at the Fifth ASTM-EURATOM Symposium on Reactor Dosimetry. Present predictions of fast neutron propagation from the core to the barrel and the neutron pad are that C/E lies between 0.85 and 1.10 in most cases, with a tendency for low values at great distance from the core and at high neutron energies.

The integral mode NRE results for VENUS, which are generally more sensitive to lower energy neutrons, reveal a more complex pattern. Integral mode scanning has been completed on six NREs irradiated in VENUS. Figure 9 shows the locations of these NRE in the VENUS mockup. Actual integral mode scanning results are given in Tables 11-16, which include C/E ratios obtained relative to MOL calculations. There is excellant agreement at core center, but serious disagreement outside the core proper.

A comparison of these results with the irradiation locations of these NRE in VENUS (see Figure 9) discloses that these discrepancies are correlated with distance for the sharp corner at the boundary of the core fuel at roughly 27°. The NRE closest to this corner, namely the NRE irradiated in the outer baffle at 24°, passes the highest C/E ratios (see Table 12). The next nearest NRE to this corner is the one located in the barrel at 21°; Table 14 shows that these C/E ratios are also high, although the J-values are somewhat improved over those of Table 12. The C/E ratios generally improve as the NRE location becomes more distant from this fuel corner. In fact, the C/E values obtained in the pad at 21° (see Table 16) agree with the above mentioned tendency for C/E values to be low at great distance from the core. Since the NRE experimental technique obviously does not change from location-to-location and excellant agreement is obtained at core center (see Table 11), one can conclude that calculations of these lower energy groups become more difficult as this fuel corner is approached. Comparison between measured and calculated gamma-energy-deposition values is still preliminary; further actions are needed to produce final experimental results.

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Cross section measurements in the ^{235}U fission spectrum neutron field at the BR1 reactor were also reported (Gi84). Microcalorimetry work in the BR1 gamma standard field has also been accomplished by J. Mason. These facilities are used as benchmark referencing fields for the VENUS measurements.

2.4.4.2 United Kingdom NESDIP Power-Reactor Ex-Vessel Cavity Benchmark

The Nestor Shielding and Dosimetry Improvement Program (NESDIP) was successfully lauched in 1983. A detailed description of the NESDIP facility at AEEW can be found in Ref (Au82,Au82a,Au83,Mc84). NESDIP efforts have been divided into three formally scheduled phases that are discussed below. Phase I (the PCA 12/13 Replica Experiments) of the program has now been completed, and an AEEW report fully detailing the experiments has been published, see Section 2.1.2. The reader is referred to this UK/NRC report and Refs (Mc84,Mc84i) for discussion of the results and conclusions.

Phase II of NESDIP, the lateral extension of PCA replica to 2-m^2 shields began in November 1983. This phase will be an AEEW responsibility in order to verify the extension of the original LWR-PV simulator assembly to this much larger size PV simulator.

Phase III of NESDIP, which involves the simulations of actual commercial LWR cavity configurations tailored to the requirements of the NRC and US utilities and vendors, is now scheduled to begin in the fall of 1985. This will be the subject of a formal agreement between NRC and AEEW since considerable US participation is entailed. In addition, timely exchange of AEEW data and analyis will be essential to meet NRC schedules. In return, AEEW has asked NRC to supply sufficient technical data including source terms to enable AEEW to calculate the γ and neutron fields within an actual PWR cavity (probably at H. B. Robinson, a Westinghouse plant operated by Carolina Power and Light Co.). Until this agreement is signed, specifics of the Phase III experiments can not be further detailed.

2.4.4.3 French DOMPAC PWR Pressure Vessel and Surveillance Capsule Benchmark

The DOMPAC dosimetry experiment is an irradiated PWR-PV and surveillance capsule sumilation performed in the pool of the TRITON reactor (Fortenay-aux-Roses). It was designed for radiation damage characterization inside the vessel (neutron spectrum variation) and a surveillance capsule located behind a simulated "thermal shield" of a reference PWR of the Fessenheim 1 (900-MW) type. A detailed description of the DOMPAC test facility can be found in Ref (A183).

Passive $^{58}Ni(n,p)$,$^{54}Fe(n,p)$ and $^{63}Cu(n,\alpha)$ RM and graphite (GAMIN) and tungsten (W) DM dosimetry measurements were performed in DOMPAC at ambient temperatures (50 to 100°C). ANISN (100-group) transport and SABINE (26-group) computations were performed for the design of DOMPAC, and the method of spectral indices was used to readjust the DOMPAC design to represent the actual water and steel configuration of Fessenheim.

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The experimental RM-Ni derived flux level results were found, generally, to be in good agreement with calculated gradients inside the vessel. A 3D Monte-Carlo (TRIPOLI-code) computation has provided validation of the experimental GAMIN and W damage fluences. It also indicates a lower effective damage threshold (0.3 MeV) than expected from the theoretical iron displacement model (0.45 MeV), which also implies weaker neutron damage attenuation inside the vessel. The damage gradient in the PV wall, evaluated experimentally by tungsten DM dosimetry, is however entirely consistent with that computed using steel damage models (iron dpa or probable zones).

2.4.5 Fifth ASTM-EURATOM International Symposium on Reactor Dosimetry and London LWR-PV-SDIP Meeting

The EURATOM and ASTM program committees successfully completed all of the necessary activities associated with implementation of the Fifth ASTM-EURATOM Symposium held in Geesthacht, Federal Republic of Germany, September 24-28, 1984. An ASTM-EURATOM program committee meeting was held in conjunction with the Reno, NV, ASTM meeting in January 1985, to initiate planning for the next symposium, tentatively scheduled to be held in San Antonio, TX, in April 1987. The symposium is scheduled to be hosted by SwRI, with E. D. Norris as the principal contact. Minutes of the 14th LWR-PV-SDIP meeting, which was hosted by RR&A and held at the Institute of Civil Engineering, London, England, October 1-5, 1984, were completed and distributed to LWR-PV-SIDP participants in December 1984.

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3.0 BENCHMARK REFERENCING AND DOSIMETRY CALIBRATION FOR RESEARCH AND POWER REACTOR TEST AND SURVEILLANCE PROGRAMS

The following sections update and provide new information about NBS and other LWR-PV-SDIP work related to benchmark referencing and dosimetry calibration activities for research, test, and power reactor studies.

3.1 CERTIFIED FLUENCE STANDARDS

A chief objective of the NBS effort is to achieve benchmark referencing of RM dosimetry measurements through the distribution and evaluation of Certified Fluence Standards. These standards are threshold and fission-type neutron sensors irradiated in a standard neutron field to a well-defined fluence.

Table 17 identifies 56 certified fluence standards prepared in the NBS ^{235}U Standard Neutron Fission Spectrum for the LWR-PV-SDIP during FY83 and FY84. The FY83 information is included because many of the standards are not evaluated for 9 to 12 months after their distribution, so FY83 information is current information on this time scale. Previous fluence standards were issued to cover PCA irradiation measurements, as reported in NUREG/CR-1861 (Mc81b) and NUREG/CR-3318 (Mc84a), and to round-robin test the $^{58}Ni(n,p)^{58}Co$ reaction; 25 irradiated nickel fluence standards were distributed and the results were reported in (Mc84j) at the 5th ASTM/EURATOM Symposium in Geesthacht, Germany, September 1984. The current (Table 17) fluence standards have been issued to cover experiments dealing with the development of power-reactor, surveillance-dosimetry benchmarks (e.g., PSF/SDMF, NESDIP and VENUS) and to support the cooperative efforts with nuclear reactor vendors to develop generic operating power reactors as combined dosimetry and metallurgy benchmarks (e.g., Maine Yankee [Combustion Engineering Designed Plant], H. B. Robinson [Westinghouse Designed Plant], and Davis Besse - Unit I [Babcock and Wilcox Designed Plant] as well as Arkansas Nuclear One - Unit I [Babcock and Wilcox Designed Plant] and Arkansas Nuclear One - Unit II [Combustion Engineering Designed Plant].

3.1.1 Benchmarking 4th SDMF Dosimetry and Round-Robin Support for Development of a Special Research Material (SRM)

As indicated in Section 2.3.1.3, NBS-certified neutron fluence standards have been sent to HEDL, KFA, Jülich, and W-NTD to benchmark reference RM sensor counting with dosimeters from the 4th SDMF irradiation. The nuclear reactions involved are $^{238}U(n,f)FP$, $^{58}Ni(n,p)^{58}Co$, $^{46}Ti(n,p)^{46}Sc$, and $^{54}Fe(n,p)^{54}Mn$. The latter two reactions were induced in pure iron and nickel foils as well as in a nickel-iron alloy containing 33.6% nickel. This alloy, with its certified nickel content, is NBS-SRM-1158. It will also be distributed, with a different number, as a Certified Fluence Standard SRM in the near future. The unirradiated alloy may be used as both a slow and fast neutron fluence monitor. The radioactive fluence standard (alloy) may be used to calibrate dosimetry methods and to determine the sensitivity of analytical methods to nickel "contamination" radiations in iron and steel samples from reactors. The gamma spectrum from an irradiated SRM alloy foil is shown in

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Figure 10. Even with a relatively high-resolution GeLi detector, proper background substraction for the iron 835-keV gamma peak, in the presence of the cobalt 811-keV peak, can be a problem.

3.1.2 Certified Fluence Standards for Evaluating ^{137}Cs Radiometric Analysis

The last irradiation in Table 17 is seen to be for 150 hours for a fluence of 10^{16} n/cm > 1 MeV to produce the first, ever, certified fluence standards containing conventionally measurable amounts of ^{137}Cs. This isotope with its 30-year half-life is the fission product most used to determine the number of fissions in any fissionable isotope. Three such ^{137}Cs standards were produced and initially distributed to HEDL, EG&G-Idaho (and NBS), who will measure radioactivity from a number of other fission products as well as ^{137}Cs. Subsequently, the ^{137}Cs fluence standards will be circulated internationally, but first to laboratories with direct interests in the NRC LWR-PV-SDIP effort.

3.2 EVALUATIONS OF POWER REACTOR SURVEILLANCE PHYSICS-DOSIMETRY EFFORTS

3.2.1 Maine Yankee

The Maine Yankee cavity dosimetry for the first, low-enrichment peripheral-fuel cycle has been removed. D. McGarry of NBS was on site in Wiscasset, Maine from 18 to 26 of May 1984 to assist H. Jones of Maine Yankee Atomic Power Company (who is responsible for the Plant's PV surveillance program) in the dosimeter recovery and a re-installation. All dosimeters were separated and re-identified to confirm loading information. HEDL dosimeters were shipped to HEDL and the remaining dosimeters (from Combustion Engineering) were repackaged for shipmenat to W-NTD (S. Anderson).

Figures 11, 12 and 13 give details of the CE dosimetry removed from the 0°, ex-vessel monitor position. Note that the aluminum wire wraps were installed (by Jones and McGarry) at Maine Yankee before installation to ensure adequate separation of the bare and cadmium-covered dosimetry packages (see Figure 12 and Figure 13).

Additional CE dosimeters, but in rectangular-aluminum HEDL-type packages, were re-installed on the original CE holder (central) rod and remounted at 0° for irradiation during the current cycle.

3.2.2 H. B. Robinson

Radiometric analyses of the ex-vessel dosimetry irradiated in Cycle 9 has been accomplished by HEDL. HEDL SSTR analysis is in progress. The internal, advanced replacement surveillance capsule dosimetry has been returned to HEDL for analyses. Correlation of internal and ex-vessel, or cavity, measurements will provide valuable information regarding verification of calculational techniques for low leakage PWR cores. There were no NBS dosimeters in either of the above-mentioned sets. Fluence standards have been supplied to HEDL for benchmarking the RM counting. All of this work is being coordinated by S. Grant of Caroline Power and Light, the utility representative responsible for the plant's PV surveillance program.

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3.2.3 McGuire-1

Preliminary results from the McGuire-1 cavity experiments have been reported at the 5th ASTM-EURATOM Symposium. These were obtained from a string of RM holders removed in 1982. Four additional strings, from cavity locations both within and between azimuthal regions covered by the reactor neutron pads and covering a range of 2 m in elevation, were retrieved in 1984 and shipped to the University of Arkansas. At the same time, an in-vessel surveillance capsule was retrieved for analysis by W-NTD (S. Anderson). NBS fluence standards have been prepared for exchange between the two counting laboratories together with the dosimetry materials. Calculations by University of Missouri-Rolla, W-NTD and ORNL are expected to provide a similar test of the in- and ex-vessel physics-dosimetry, which has been completed in ANO-2.

3.2.4 Turkey Point 3

Advanced cavity dosimetry was installed by W-NTD in the reactor cavity of Turkey Point Unit-3 in December 1983. The installed cavity dosimetry includes both radiometric monitors (HEDL-ILRR foils and NBS paired uranium detectors). J. Sun of Florida Power and Light is responsible for the coordination of this joint NRC-HEDL-NBS-W-NTD physics-dosimetry surveillance program verification effort.

3.2.5 Arkansas Nuclear One - Unit II

Experiments similar to those in ANO-1 commenced in 1979. Four detector wells have been used for cavity dosimetry, with radiometric monitors at several elevations and axial profiles covering the core region and beyond the nozzles, above the installed cavity shield. Seven cavity experiments at various stages of three fuel cycles have been completed and, as in Unit I, cavity measurements are to be continued. An internal surveillance capsule was removed in 1982, from which dosimetry materials have been analyzed by Battelle Columbus Laboratories and by the University of Arkansas. Both of these laboratories have also counted neutron fluence standards supplied by the NBS.

3.3 COOPERATIVE RESEARCH BETWEEN NBS AND SCK/CEN RELATED TO THE NRC LWR-PV-SDIP

3.3.1 NBS Participation in the VENUS Experiment

Extensive dosimetry measurements have been completed for the Belgian VENUS Program for core source to PV wall fluence verification. J. Grundl, D. Gilliam and D. McGarry of NBS were on site in Mol, Belgium (in 1983 and 1984) to assist with the subject experiment and to verify benchmark field calibrations of all the active neutron dosimetry used therein (see Section 3.3.2.3). In VENUS, prototype experimental data were obtained for testing calculational methods and models in the vicinity of nuclear reactor fuel corners and core edges, see Section 2.4.4.1.

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Significant analytical effort was required during 1984, some of which is carried into FY85, to digest the experimental information described in the following sections.

3.3.2 Intercalibration of the Belgian and U.S. Standard ^{235}U Fission Spectrum Irradiation Facilities

3.3.2.1 U.S. ^{235}Cavity Fission Source

The U.S. Standard ^{235}U Fission Neutron Spectrum is at the National Bureau of Standards in Gaithersburg, MD. It operates in the center of a 30-cm diameter spherical cavity located in the graphite thermal column of the NBS Research Reactor. For this reason, the facility is frequently called the Cavity Fission Source (CFS). The upper view in Figure 14 shows, in detail, the CFS assembly containing neutron sensors and the lower view shows the CFS's location within the thermal column cavity. The CFS consists of two disks of ^{235}U metal (16 mm diam x 0.13 mm thick) placed above and below a cylindrical cadmium pill box 0.076 cm thick. This assembly encloses, for irradiation, approximately six passive neutron sensors (nominally 1.27 cm diam x 0.025 cm thick). The neutron sensors are held in the center of the assembly by light-weight aluminum pieces.

3.3.2.2 Belgian ^{235}U Cavity Fission Source

Figure 15 shows the ^{235}U CFS assembly at the SCK/CEN Laboratory, Mol, Belgium. This assembly incorporates a 100-cm diam cavity as opposed to the 30-cm diam cavity for the NBS ^{235}U field. The significance of the larger cavity is that there are substantially fewer low-energy (wall returned) neutrons in the Belgian field. Also, the larger source permits the exposure of an NBS fission chamber. This chamber cannot be used in the smaller volume NBS CFS. The Belgian ^{235}U Standard Field operates in the graphite thermal column of the BR-1 reactor at the SCK/CEN Laboratories.

3.3.2.3 Verification of Certified Fluences in the Belgian ^{235}U Standard Field

The absolute source strength, and therefore the fluence rate, in any CFS is dependent on the reactor power level. Consequently, a fluence monitor that is independently calibrated against both an absolute source strength and the reactor power level is required for each irradiation. The independent calibration of the Belgian ^{235}U neutron field was confirmed by NBS. The fission rate, per watt of reactor power, in the Belgian field was compared with the fission rate of ^{239}Pu during a certified flux-density irradiation in the NBS ^{252}Cf Standard Fission Spectrum.

The source for the NBS ^{252}Cf (spontaneous fission) neutron field is a point source bead of $^{252}CfP_2$ in a light-weight aluminum and stainless steel capsule. The source is suspended 1.6 m from the nearest reflecting surface in a low-scatter environment. The free-field fluence rate for such a ^{252}Cf source is then just a function of source strength, distance, time and a few small geometry-dependent scattering corrections.

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Since the ratio of the $^{239}Pu(n,f)$ spectrum-averaged cross sections in the ^{235}U and ^{252}Cf fission spectra is unity, to within several tenths of a percent, the fission rate per monitored watt of reactor power in the Belgian ^{235}U spectrum is measured with the same ^{239}Pu deposit, in an NBS fission chamber, as was used in the NBS ^{252}Cf measurement.

3.3.2.4 Recalibration of the NBS ^{235}U Standard Fission Spectrum Against the Belgian ^{235}U Fission Spectrum

It should be noted that if it were not for the fact that the NBS ^{235}U CFS is too small to accept an NBS fission chamber, calibration of its flux density could be accomplished with a ^{239}Pu deposit in a fission chamber in the same manner that Belgian ^{235}U field was calibrated. However, such is not the case. The method of calibration is to utilize a radiometric dosimeter sufficiently small in size to fit into the NBS CFS but with a reaction probability large enough to provide radioactivity sufficient for counting after a few days of irradiation in the standard neutron field.

Historically, the reaction selected was the $^{115}(In(n,n')^{115m}In$ reaction and the flux-density calibration for the NBS ^{235}U field was transferred from the NBS ^{252}Cf field. In 1983-1984, the transfer was accomplished from the Mol ^{235}U field by the $^{58}Ni(n,p)^{58}Co$ reaction. The change was made because the 4.5-hour half-life of the indium reaction is too short to transfer from Belgium, while the reaction rate probability of the nickel reaction is too low for efficient transfer from the ^{252}Cf field. Figure 16 summarizes the details of both the "old" and "new" calibrations. At present, there is about a 1.5% discrepancy between the two methods. This is, of course, not a problem for bencmkark referencing LWR-PV-SDIP dosimetry.

All of the above activities as related to standard neutron fields have been documented and reported in the proceedings from the 5th ASTM/EURATOM Symposium held in Geesthacht, Germany, September 1984. The information, and its significance as a bases of reference for dosimetry accomplished under the NRC LWR-PV-SDIP, was presented at Geesthacht in a poster paper.

3.4 NBS INTERACTIONS WITH PSF/SDMF AND PCA BENCHMARK IRRADIATION EXPERIMENTS

3.4.1 Certified Fluence Standards

As indicated in Table 17, during FY83, FY84 and so far in FY85, approximately 20 certified fluence standards have been distributed to HEDL, CEN/SCK, ORNL, W-NTD, and KFA (Jülich) to assist in benchmark referencing radiometric dosimetry for the various PSF/SDMF experiments. Table 18 gives an accounting by dosimeter types and reactions.

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3.4.2 ORR-SDMF

NBS assisted HEDL and ORNL in dosimeter logistics, planning and coordinating the November 1983 4th SDMF: a nominal 18-day irradiation of selected radiometric (RM) dosimetry (from HEDL, NBS and KFA), SSTRs (HEDL), HAFMs (RI) and damage monitors (DM) (Harwell, RR&A) in the 4/12 SSC configuration. Fluence standards (already identified in Table 18) were sent to HEDL and KFA. The NBS paired-uranium detectors (PuDs) have been shipped to HEDL for radiometric analyses. Significant heat plus water damage (apparently) was suffered by the PuD in the SSC location. Probably no data will be available from that dosimeter.

3.4.3 Special Standard Neutron Field Irradiation

Several ^{252}Cf and ^{235}U standard neutron field irradiations were conducted to examine various aspects of SSTR and HAFM methodology. Except for the very (ultra) light-weight SSTR fissionable deposits, these irradiations have confirmed assigned masses for the HEDL deposits and have demonstrated that in free fields, such as found in the vicinity of a ^{252}Cf source, both NBS fission chambers and SSTRs yield the same fission rates to within $\pm 1\%$ (1σ).

NBS is actively engaged in the present controversy where HAFM measurements in a variety of benchmark and test region neutron fields suggest discrepancies with the ^{10}B and $^{6}Li(n,\alpha)$ cross sections.

3.5 COMPENDIUM OF BENCHMARK AND TEST-REGION NEUTRON FIELDS AND RELATED RESEARCH

3.5.1 Compendium

The subject compendium was updated and published in FY84 in NUREG/CR-3391, Vol 4 (Mc84k). Since then, an additional entry has been completed for the Intermediate-Energy Standard Neutron Field (ISNF) at NBS, and the update will be published in FY85.

In addition to the ISNF, work is progressing toward an entry for the PCA Experiments. Neutron energy spectra and physical description have been received from ORNL for the PCA 12/13 and 4/12 SSC configurations. Spectrum information for the 8/7 configuration is still required.

3.5.2 Related Research*

In the Belgian Cavity Fission ^{235}U spectrum at the BR-1 Reactor at Mol, Belgium SCK/CEN Laboratories, a series of neutron cross section measurements was made in which the neutron fluence rate was determined by a transfer

*This research is not funded by the NRC but is of interest because it has a direct bearing on the accuracy of benchmark referenced LWR-PV dosimetry and standard neutron field data in the subject compendium.

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from the NBS ^{252}Cf Fission Spectrum. The mentioned flux-density transfer had been accomplished in support of calibration measurements for the Belgian/NRC VENUS program (see Section 3.3.2.3). This provided an opportunity to verify integral cross-section measurements for important dosimetry isotopes. Fission cross sections were reported for ^{238}U, ^{235}U, ^{233}U, ^{240}Pu, ^{237}Np and ^{232}Th, as well as threshold reaction cross sections for ^{115}In(n,n') and the ^{58}Ni(n,p) reactions. The data were reported at the 5th ASTM/EURATOM symposium by D. Gilliam et al. (Gi84). The results are summarized in Tables 19 and 20.

Both the US (NBS Laboratory) and the FRG (PTB Laboratory) have complete cross-section measurements in the ^{235}U fission spectrum at Mol, Belgium. The PTB results were presented by W. Mannhart at the 5th ASTM/EURATOM Symposium. Agreement for common isotopes is very good. Furthermore, within the experimental uncertainties, the results still agree with the measurements made at Mol, by CEN/SCK, NBS, and PTB.

3.5.3 Scattering in the Cavity Fission Source

The ^{235}U spectrum of the NBS CFS is created inside of a small volume between two fission disks. Compared to the complexity of nuclear power reactors, the CFS is a relatively simple structure. As such, it can be mathematically modeled in Monte Carlo calculations and can be examined for effects of interactions on the purity of the fission spectrum.

The Monte Carlo investigations, carried out for NBS by P. Sorenson (LANL), reveal net spectral perturbations on the order of 5%; primarily, because of inelastic scattering in uranium and aluminum. The energy shift in the fission spectrum is shown in Figure 17. This figure is the spectrum calculated at a particular fluence-monitor sensor within the CFS. By integrating the perturbed spectrum over the cross section of the pertinent neutron sensor material, one determines the effect on a particular isotopic reaction rate. Such an effect may be negative or positive. For example, a negative result will be obtained whenever inelastic scattering decreases the neutron energy below the threshold of a reaction. Figure 18 graphically represents the trends in such perturbations as a function of the approximate low-energy thresholds. The five data points (open circles) are, from left to right, for the reactions ^{237}Np(n,f), ^{115}In(n,n'), ^{58}Ni(n,p), ^{27}Al(n,α) and ^{63}Cu(n,α).

3.6 QUALITY ASSURANCE OF AN INVENTORY OF SENSOR MATERIALS FOR SURVEILLANCE DOSIMETRY

3.6.1 New Supply of NBS ^{238}U Radiometric Dosimeters

Negotiations are underway with the Belgian, Geel Laboratory (Van Audenhoven) to fabricate, from NBS-SRM-002 uranium oxide powder (U_3O_8), a supply of quality controlled ^{238}U dosimeters. The mentioned SRM material has a 0.01755 at.% ^{235}U content. The final form, pressed powder or metal foil, on the material for encapsulation has not yet been decided. Dosimeter geometry will be a disk nominally 13 mm diam by 0.15 to 0.40 mm thick. Specifications require elemental weight of uranium in each dosimeter be known and that any residual oxide or other impurities that would affect the gravimetric analysis at the 0.5% accuracy level should also be known.

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3.6.2 Cobalt-in-Copper Analyses

Because of the relatively large sensitivity of cobalt to low energy neutrons and the small fast neutron cross section of the $^{63}Cu(n,\alpha)$ reaction, the cobalt impurity of copper dosimeters must be taken into account.

NBS Analytical Chemistry Division (R. Fleming) has established that reactor experiment copper foils have a net cobalt content of 43 $\pm$ ppb(ng/g) of cobalt -- the apparent limit of detectability for the radiometric method used is $\sim$15 ng/g of cobalt established by looking for evidence of cobalt in a "100% pure" copper sample. This is 15 ng/g limit subtracted from all observed cobalt contents to derive the quoted net contents.

3.7 ASTM STANDARD GUIDE E706(IIE) FOR BENCHMARK TESTING OF REACTOR-VESSEL NEUTRON DOSIMETRY

3.7.1 Status

With reference to Section 2.1.1, a fairly comprehensive draft of the ASTM E706(IIE) standard was distributed at the Williamsburg, VA ASTM E10.05 meeting in June 1984. Some comments have been received and incorporated. However, a difficult section to deal with is that of error propagation.

3.7.2 Error Assessment

Error propagation with integral detectors is complex because such detectors do not measure neutron fluence directly, and because the same measured detector responses from which a neutron fluence is derived are also used to help establish the neutron spectrum required for the fluence derivation. Involved error correlations are the result and they do not simplify when benchmark referencing is employed. Nevertheless, with judicious assumptions, tailored to the characteristics of a given experiment, it is possible to obtain a diagnosis of error sources and their approximate magnitude.

All of this is not entirely consistent with the results of least squares adjustment methods and their co-variance matrix data. Further study of the error assessment problem is, therefore, required.

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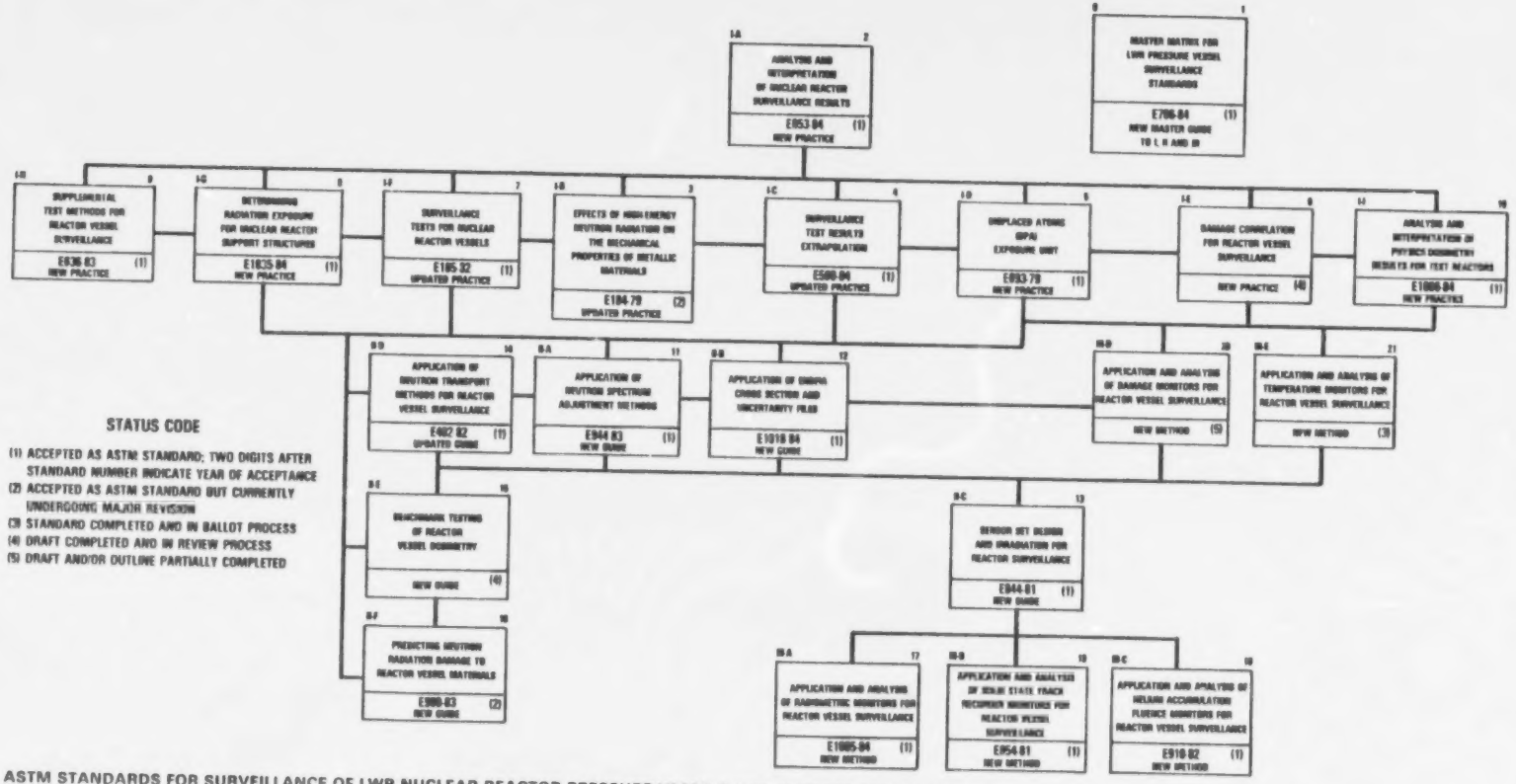
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(1) ACCEPTED AS ASTM STANDARD; TWO DIGITS AFTER STANDARD NUMBER INDICATE YEAR OF ACCEPTANCE
(2) ACCEPTED AS ASTM STANDARD BUT CURRENTLY UNDERGOING MAJOR REVISION
(3) STANDARD COMPLETED AND IN BALLOT PROCESS
(4) DRAFT COMPLETED AND IN REVIEW PROCESS
(5) DRAFT AND/OR OUTLINE PARTIALLY COMPLETED
ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS
E853-84 (1)
NEW PRACTICE
MASTER MATRIX FOR LWR PRESSURE VESSEL SURVEILLANCE STANDARDS
E706-84 (1)
NEW MASTER GUIDE TO I, II AND III
SUPPLEMENTAL TEST METHODS FOR REACTOR VESSEL SURVEILLANCE
E636-83 (1)
NEW PRACTICE
DETERMINING RADIATION EXPOSURE FOR NUCLEAR REACTOR SUPPORT STRUCTURES
E1035-84 (1)
NEW PRACTICE
SURVEILLANCE TESTS FOR NUCLEAR REACTOR VESSELS
E185-82 (1)
UPDATED PRACTICE
EFFECTS OF HIGH-ENERGY NEUTRON RADIATION ON THE MECHANICAL PROPERTIES OF METALLIC MATERIALS
E184-79 (2)
UPDATED PRACTICE
SURVEILLANCE TEST RESULTS EXTRAPOLATION
E509-84 (1)
UPDATED PRACTICE
DISPLACED ATOMS (DPA) EXPOSURE UNIT
E693-79 (1)
NEW PRACTICE
DAMAGE CORRELATION FOR REACTOR VESSEL SURVEILLANCE
NEW PRACTICE (4)
ANALYSIS AND INTERPRETATION OF PHYSICS DOSIMETRY RESULTS FOR TEST REACTORS
E1006-84 (1)
NEW PRACTICE
APPLICATION OF NEUTRON TRANSPORT METHODS FOR REACTOR VESSEL SURVEILLANCE
E482-82 (1)
UPDATED GUIDE
APPLICATION OF NEUTRON SPECTRUM ADJUSTMENT METHODS
E944-83 (1)
NEW GUIDE
APPLICATION OF ENDF/B CROSS SECTION AND UNCERTAINTY FILES
E1018-84 (1)
NEW GUIDE
APPLICATION AND ANALYSIS OF DAMAGE MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD (5)
APPLICATION AND ANALYSIS OF TEMPERATURE MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD (3)
BENCHMARK TESTING OF REACTOR VESSEL DOSIMETRY
NEW GUIDE (4)
PREDICTING NEUTRON RADIATION DAMAGE TO REACTOR VESSEL MATERIALS
E900-83 (2)
NEW GUIDE
SENSOR SET DESIGN AND IRRADIATION FOR REACTOR SURVEILLANCE
E944-81 (1)
NEW GUIDE
APPLICATION AND ANALYSIS OF RADIOMETRIC MONITORS FOR REACTOR VESSEL SURVEILLANCE
E1005-84 (1)
NEW METHOD
APPLICATION AND ANALYSIS OF SOLID STATE TRACK RECORDER MONITORS FOR REACTOR VESSEL SURVEILLANCE
E854-81 (1)
NEW METHOD
APPLICATION AND ANALYSIS OF HELIUM ACCUMULATION FLUENCE MONITORS FOR REACTOR VESSEL SURVEILLANCE
E910-82 (1)
NEW METHOD

ASTM STANDARDS FOR SURVEILLANCE OF LWR NUCLEAR REACTOR PRESSURE VESSELS AND SUPPORT STRUCTURES.

HEDL 8407 057

FIGURE 1. ASTM Standards for Surveillance of LWR Nuclear Reactor Pressure Vessels and Their Support Structures.

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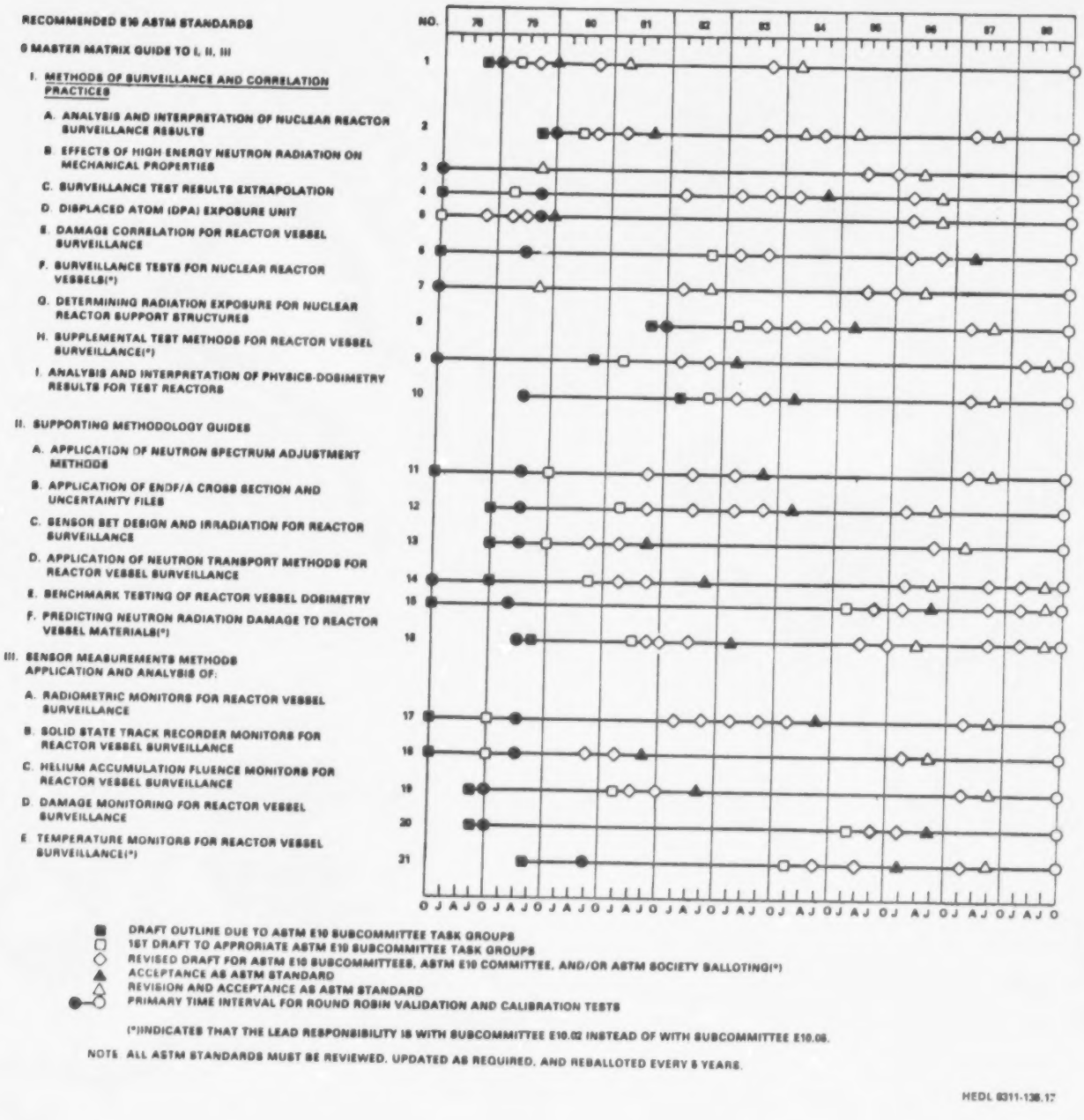
RECOMMENDED E10 ASTM STANDARDS
0 MASTER MATRIX GUIDE TO I, II, III
I. METHODS OF SURVEILLANCE AND CORRELATION PRACTICES
A. ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS
B. EFFECTS OF HIGH ENERGY NEUTRON RADIATION ON MECHANICAL PROPERTIES
C. SURVEILLANCE TEST RESULTS EXTRAPOLATION
D. DISPLACED ATOM (DPA) EXPOSURE UNIT
E. DAMAGE CORRELATION FOR REACTOR VESSEL SURVEILLANCE
F. SURVEILLANCE TESTS FOR NUCLEAR REACTOR VESSELS(*)
G. DETERMINING RADIATION EXPOSURE FOR NUCLEAR REACTOR SUPPORT STRUCTURES
H. SUPPLEMENTAL TEST METHODS FOR REACTOR VESSEL SURVEILLANCE(*)
I. ANALYSIS AND INTERPRETATION OF PHYSICS-DOSIMETRY RESULTS FOR TEST REACTORS
II. SUPPORTING METHODOLOGY GUIDES
A. APPLICATION OF NEUTRON SPECTRUM ADJUSTMENT METHODS
B. APPLICATION OF ENDF/A CROSS SECTION AND UNCERTAINTY FILES
C. SENSOR SET DESIGN AND IRRADIATION FOR REACTOR SURVEILLANCE
D. APPLICATION OF NEUTRON TRANSPORT METHODS FOR REACTOR VESSEL SURVEILLANCE
E. BENCHMARK TESTING OF REACTOR VESSEL DOSIMETRY
F. PREDICTING NEUTRON RADIATION DAMAGE TO REACTOR VESSEL MATERIALS(*)
III. SENSOR MEASUREMENTS METHODS APPLICATION AND ANALYSIS OF:
A. RADIOMETRIC MONITORS FOR REACTOR VESSEL SURVEILLANCE
B. SOLID STATE TRACK RECORDER MONITORS FOR REACTOR VESSEL SURVEILLANCE
C. HELIUM ACCUMULATION FLUENCE MONITORS FOR REACTOR VESSEL SURVEILLANCE
D. DAMAGE MONITORING FOR REACTOR VESSEL SURVEILLANCE
E. TEMPERATURE MONITORS FOR REACTOR VESSEL SURVEILLANCE(*)
NO.
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O J A J O J A J O J A J O J A J O J A J O J A J O J A J O J A J O J A J O J A J O J A J O
DRAFT OUTLINE DUE TO ASTM E10 SUBCOMMITTEE TASK GROUPS
1ST DRAFT TO APPROPRIATE ASTM E10 SUBCOMMITTEE TASK GROUPS
REVISED DRAFT FOR ASTM E10 SUBCOMMITTEES, ASTM E10 COMMITTEE, AND/OR ASTM SOCIETY BALLOTING(*)
ACCEPTANCE AS ASTM STANDARD
REVISION AND ACCEPTANCE AS ASTM STANDARD
PRIMARY TIME INTERVAL FOR ROUND ROBIN VALIDATION AND CALIBRATION TESTS
(*)INDICATES THAT THE LEAD RESPONSIBILITY IS WITH SUBCOMMITTEE E10.02 INSTEAD OF WITH SUBCOMMITTEE E10.05.
NOTE: ALL ASTM STANDARDS MUST BE REVIEWED, UPDATED AS REQUIRED, AND REBALLOTED EVERY 5 YEARS.
HEDL 8311-138.12

FIGURE 2. Preparation, Validation, and Calibration Schedule for LWR-PV and Support Structure Surveillance Standards.

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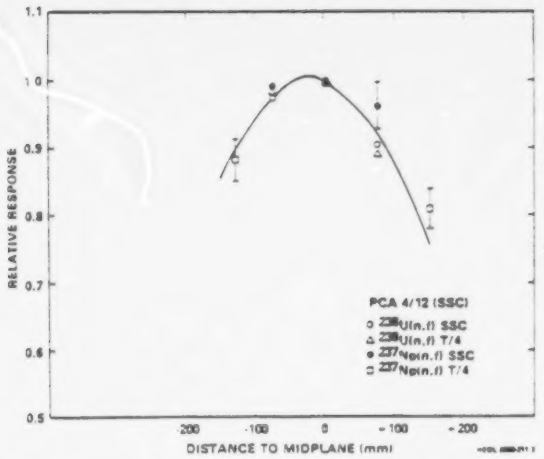

FIGURE 3. Axial Distributions of the ^{237}Np and ^{238}U Fission Rates in the PCA 4/12 SSC Configuration.

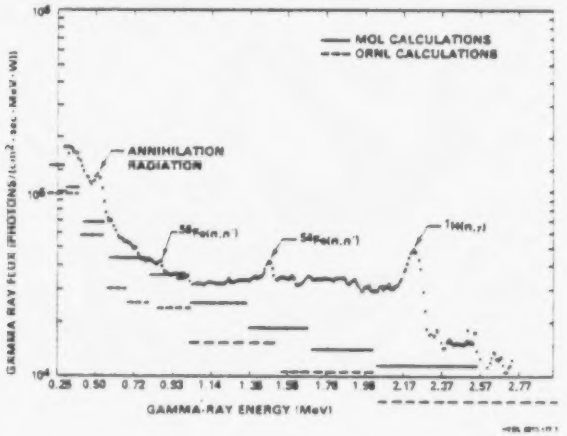

FIGURE 4. Low-Energy Gamma-Ray Continuum for the 1/4-T Location of the 12/13 Configuration as Compared with CEN/SCK and ORNL Calculational Results.

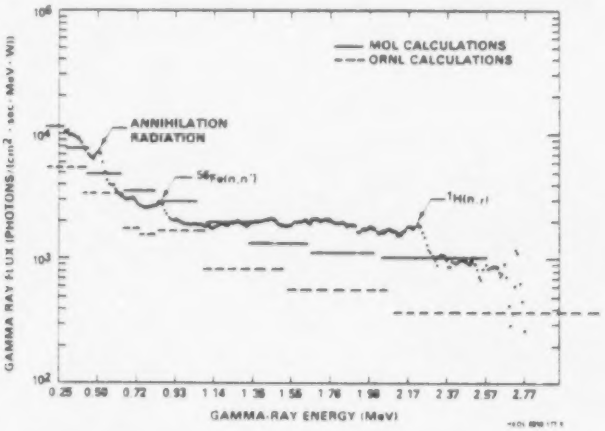

FIGURE 5. Low-Energy Gamma-Ray Continuum for the 3/4-T Location of the 12/13 Configuration as Compared with CEN/SCK and ORNL Calculational Results.

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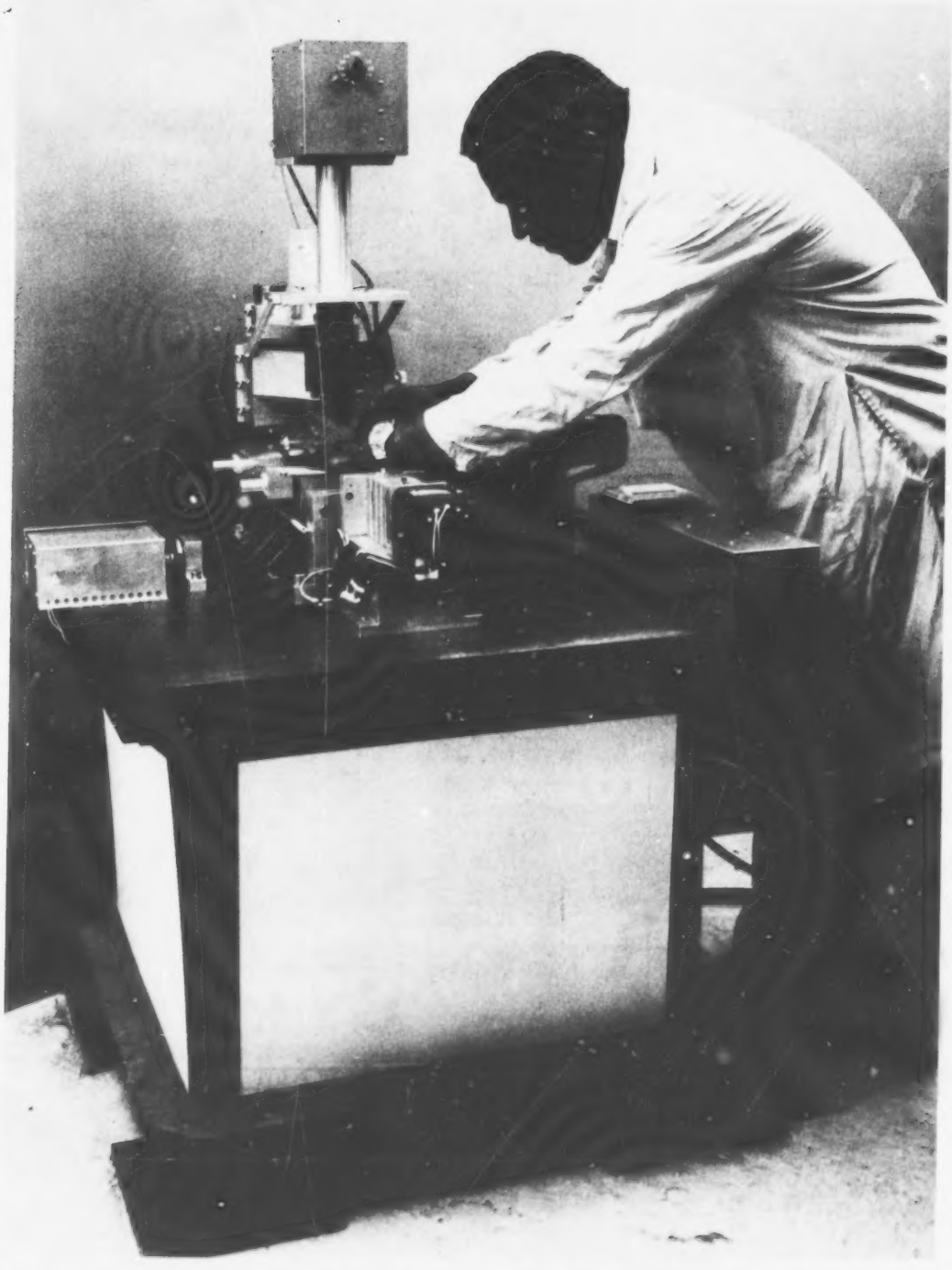

FIGURE 6. View of Hanford Optical Track Scanning (HOTS) System for SSTRs Showing Scanning Stage and Optics. Neg 8102282-4

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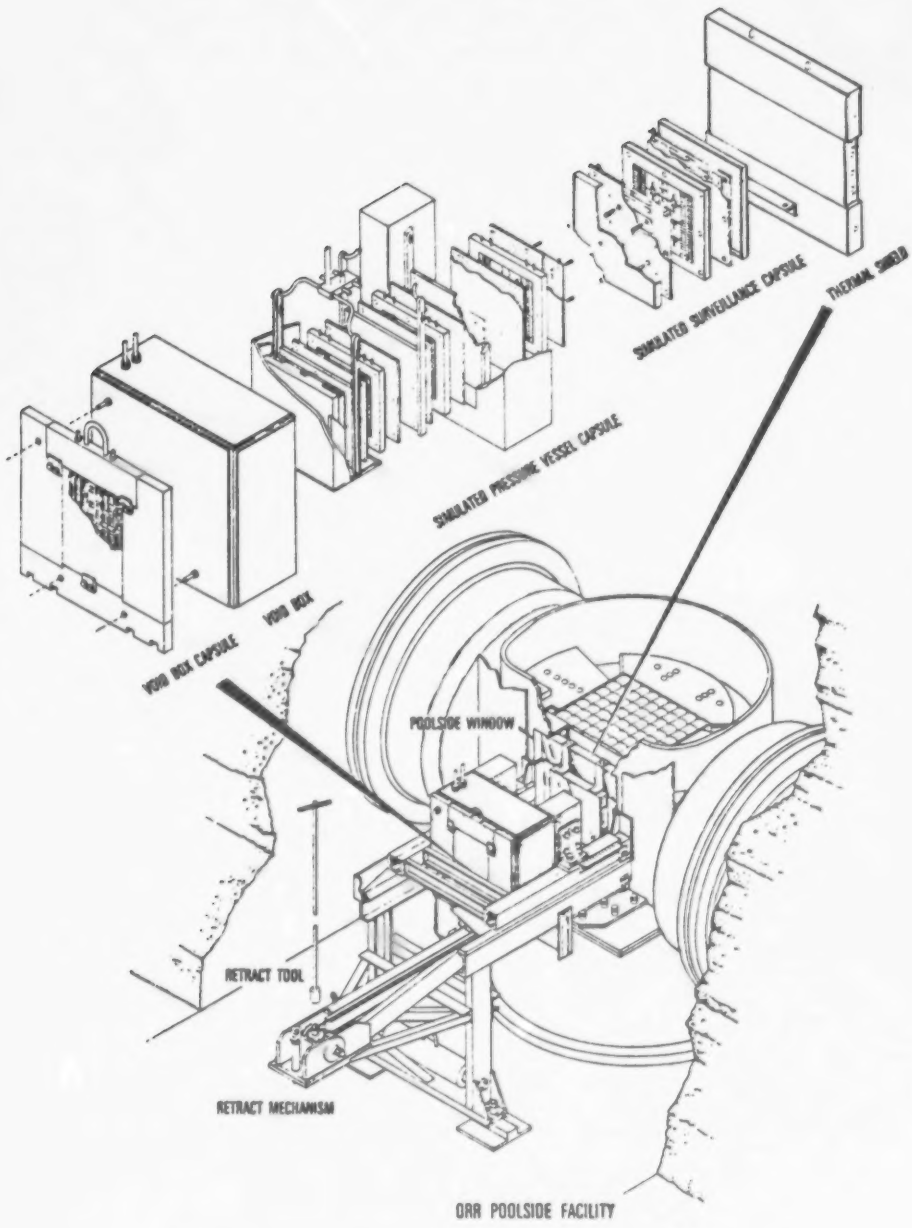

FIGURE 7. ORR-PSF Irradiation Facility.

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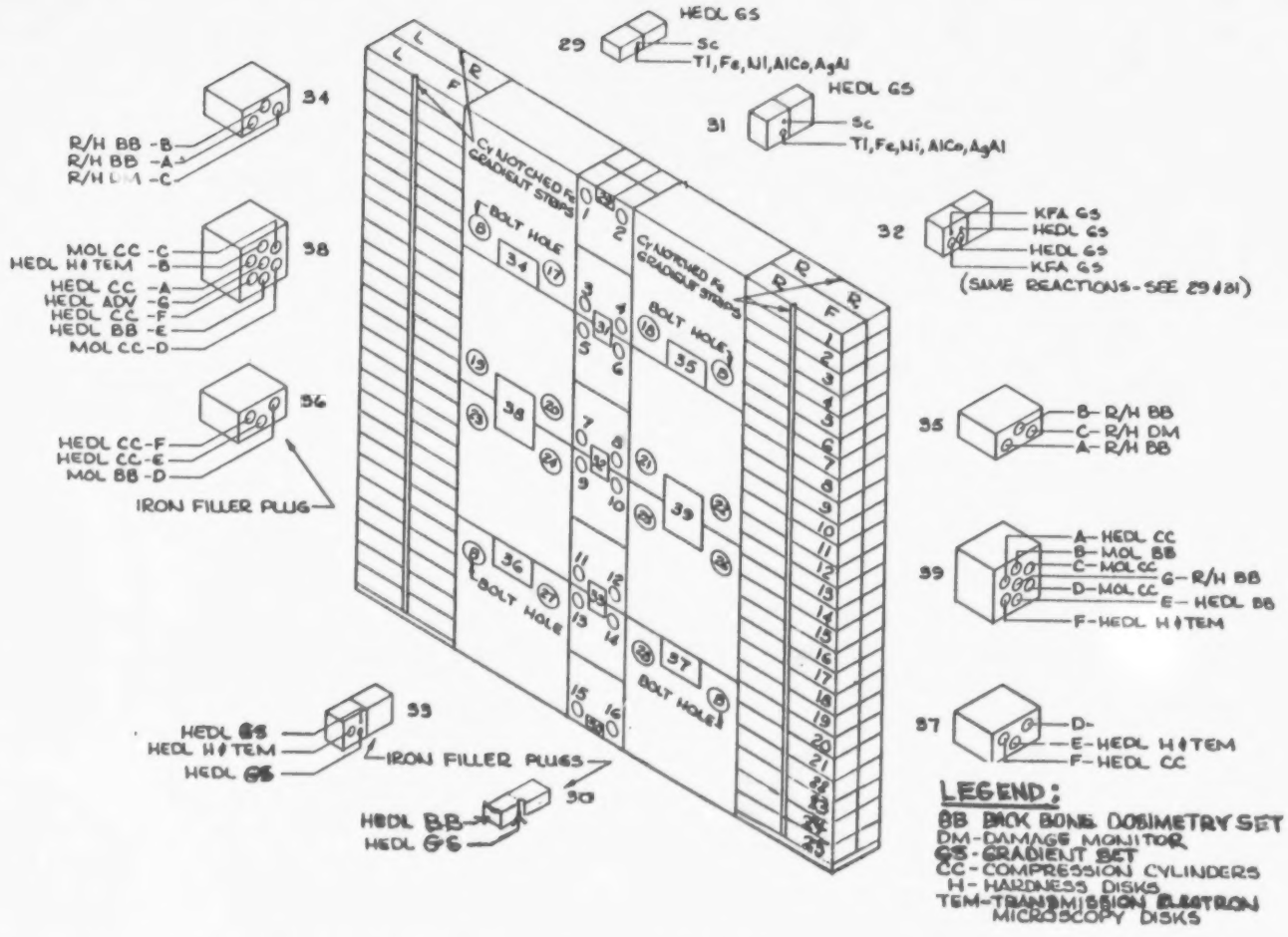

FIGURE 8. Illustration of a Typical Dosimeter and Metallurgical Specimen Assembly in the Irradiation Capsules.

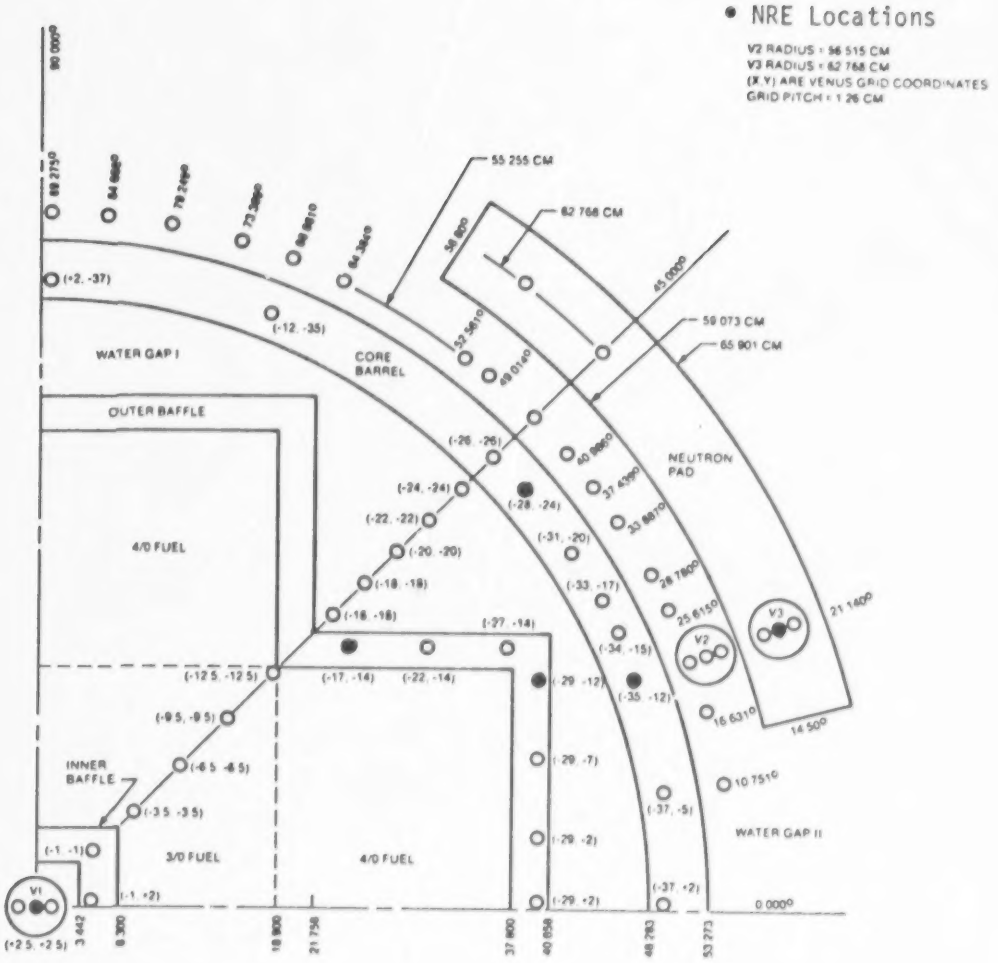

FIGURE 9. Venus Irradiation Locations of Selected NRE That Have Been Scanned in the Integral Mode.

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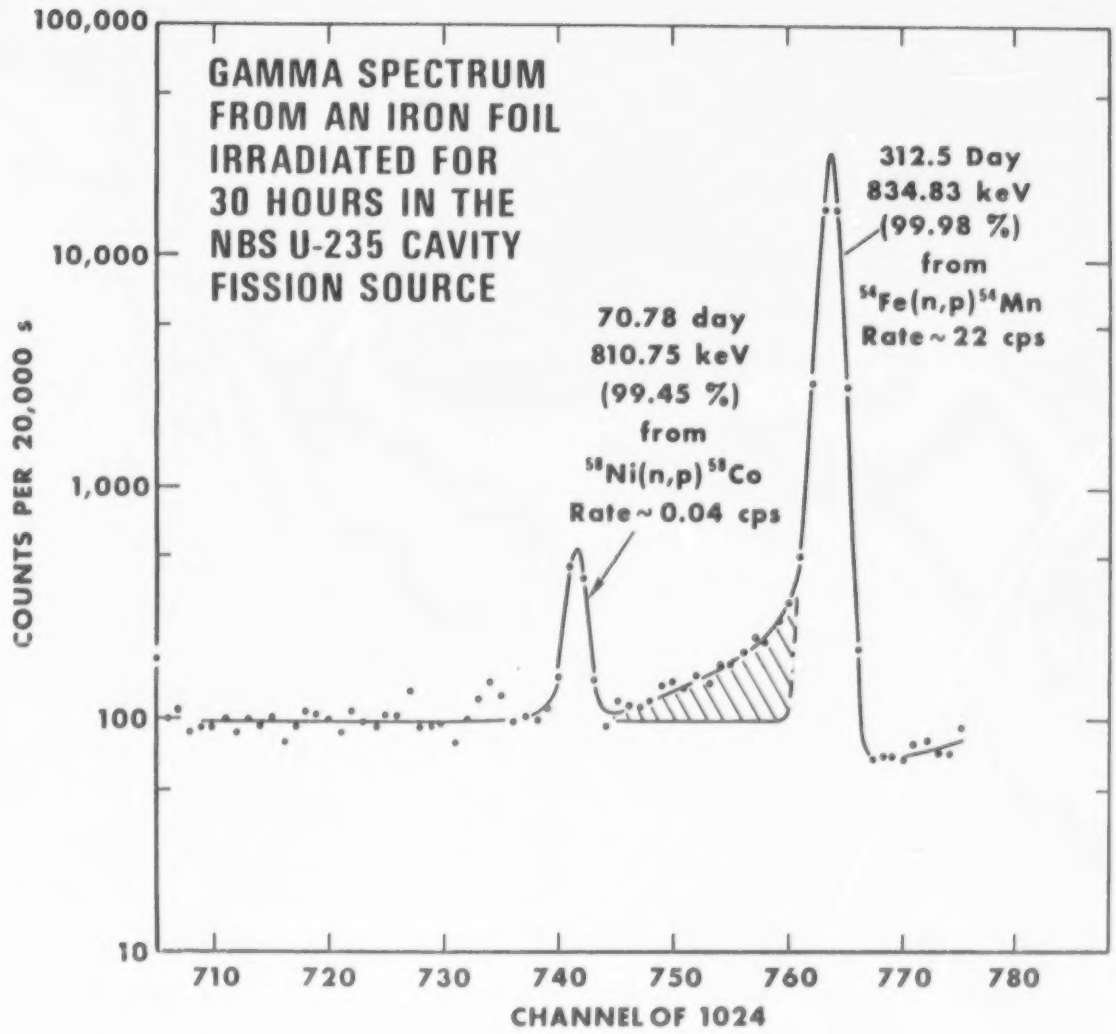

FIGURE 10. Gamma Spectrum from an Iron Foil Irradiated for 30 Hours in the NBS ^{235}U Cavity Fission Source.

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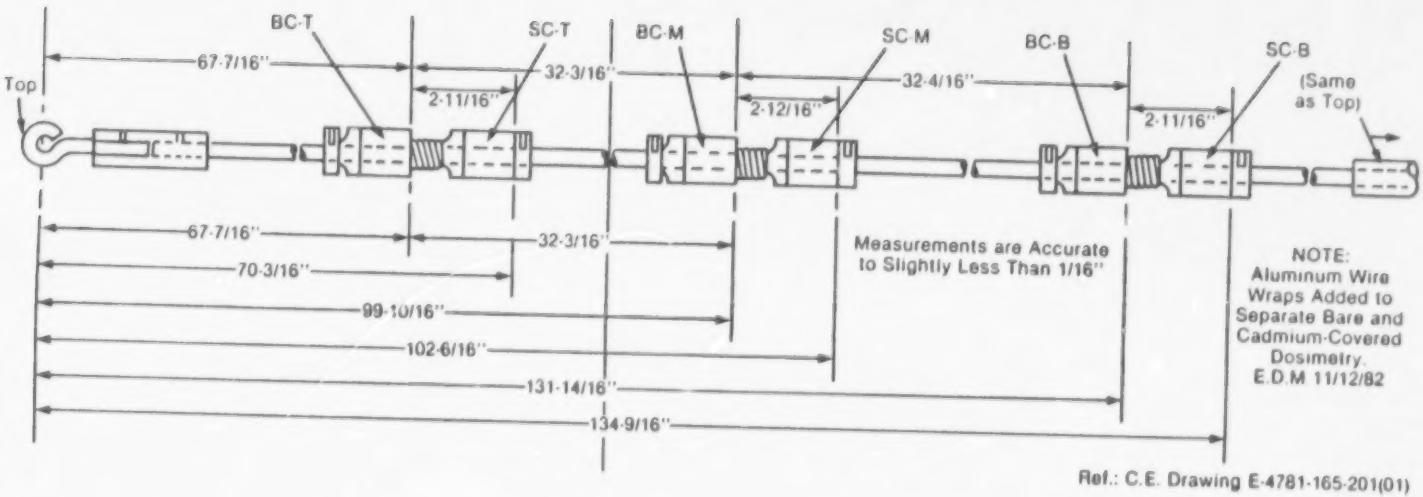

NOTE: Slight Discrepancies (± 1/16") Exist in Redundant Measurements. Principal Cause was Inability to Reproduce Top Reference Position.

FIGURE 11. Record of On Site (Maine Yankee) Changes to Combustion Engineering's 0° Ex-Vessel Flux Monitor Capsule Assembly. Dimensions shown are "as recorded" measurements. Spring-like, aluminum-wire wraps were added on site.

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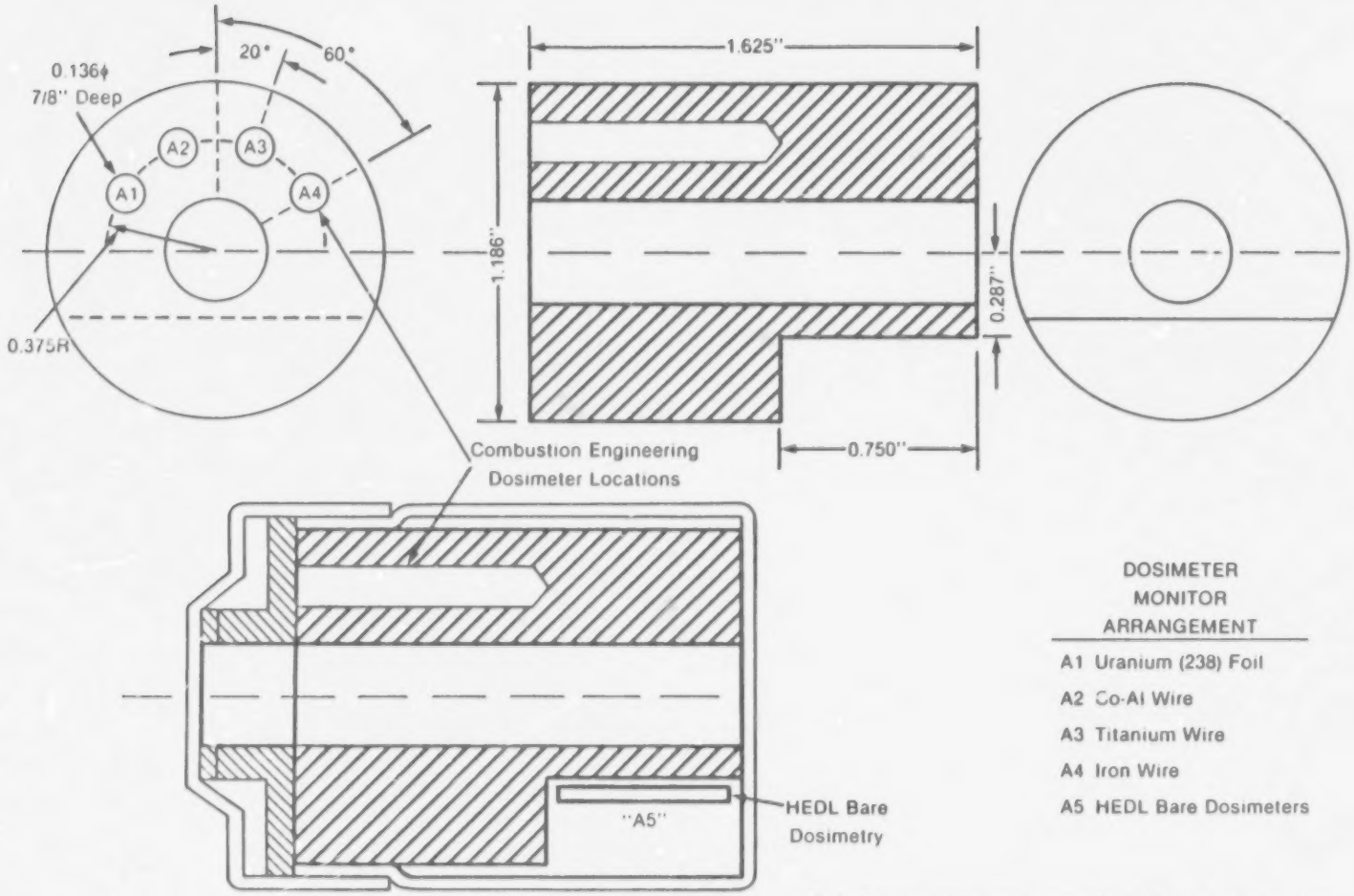
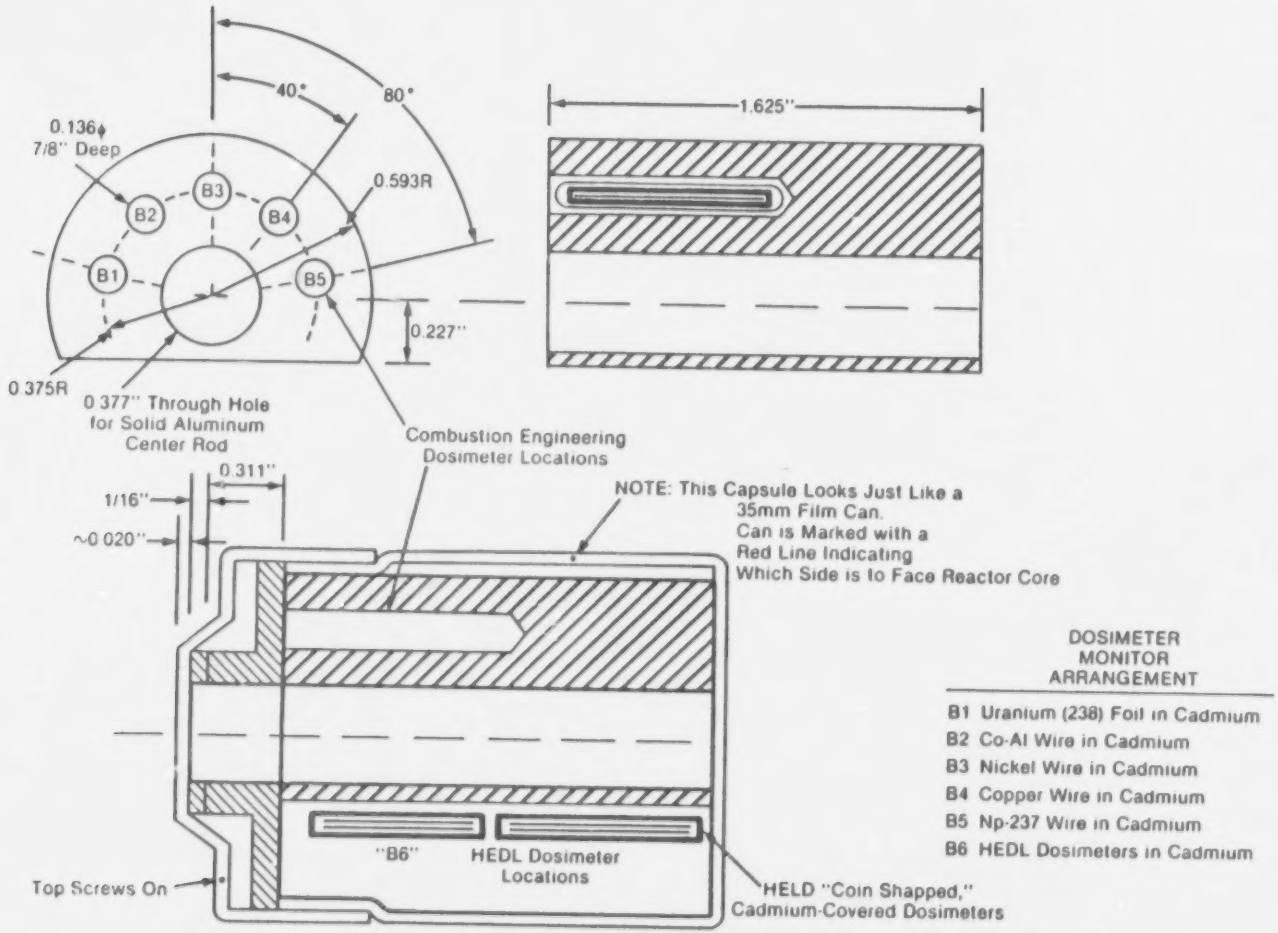


FIGURE 12. Bare Dosimetry Package for Maine Yankee 0° Dosimeters.



NOTE: The Cadmium Covers are on the Dosimeters. The Above Parts are all Aluminum.

FIGURE 13. Shielded (Cadmium) Dosimetry Package for Maine Yankee 0° Dosimeters.

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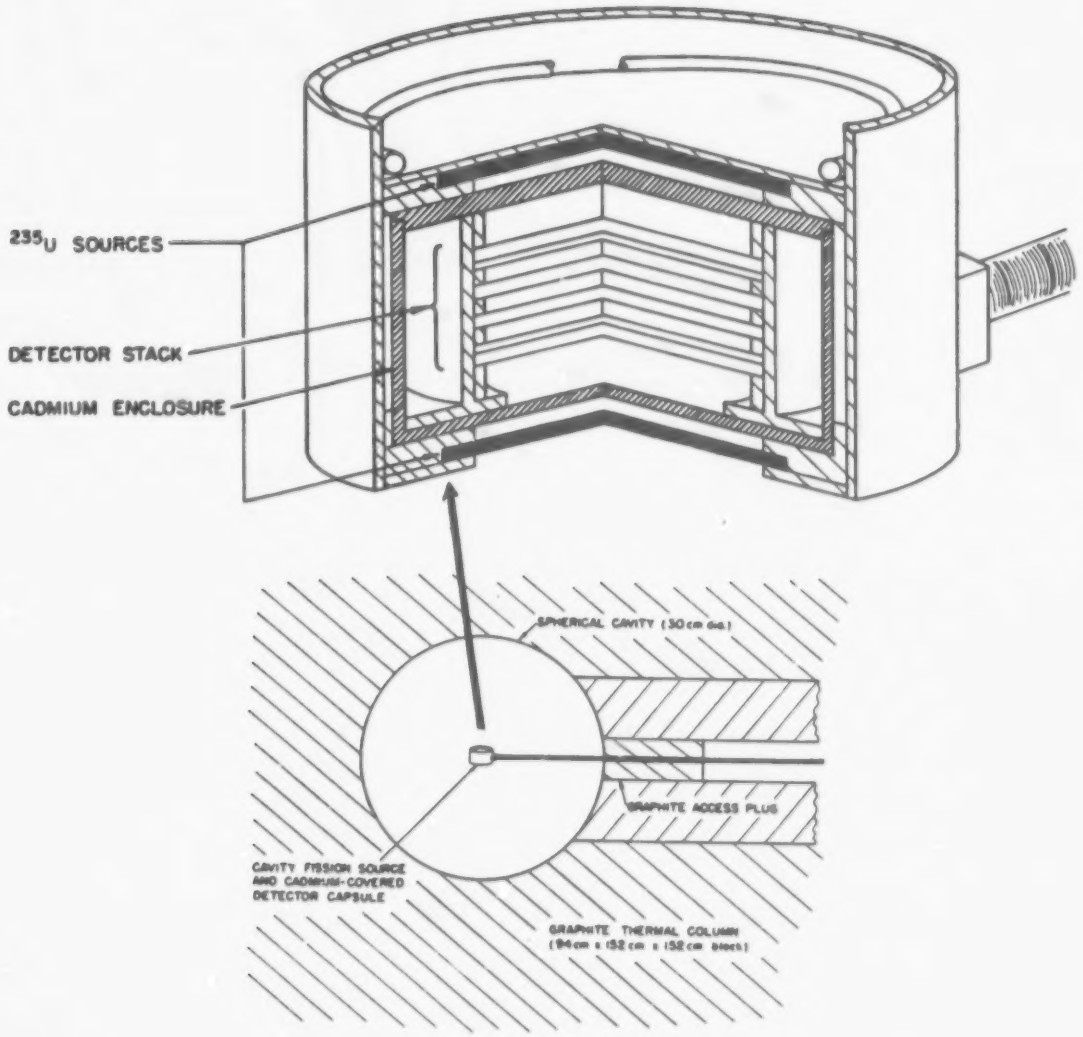

FIGURE 14. NBS Cavity Fission Source.

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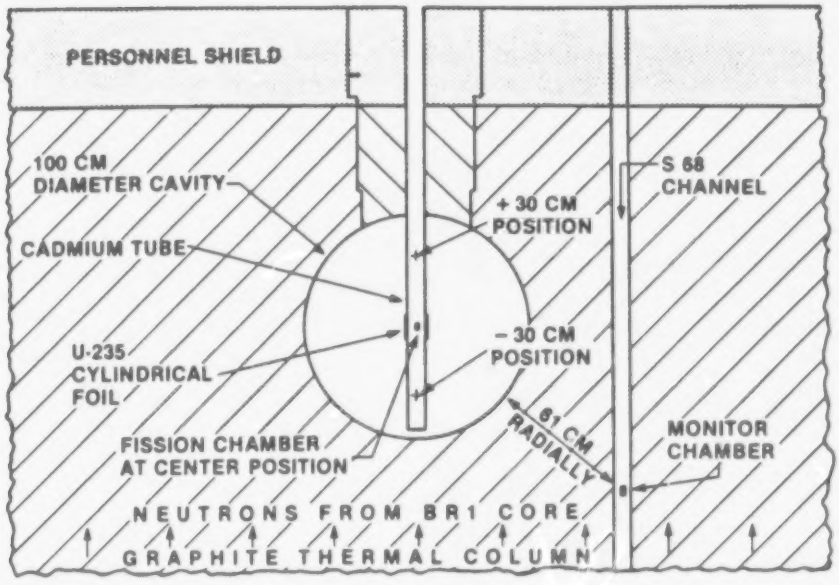
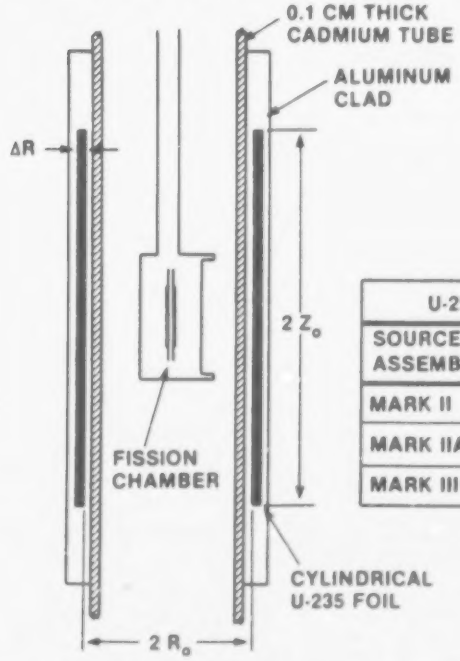


FIGURE 15a. BR-1 Cavity Fission Spectrum Standard Neutron Field Facility.



U-235 FOIL DIMENSIONS			
SOURCE ASSEMBLY NO.	2 Z_o (CM)	2 R_o (CM)	ΔR (CM)
MARK II	7.7	3.3	0.01
MARK IIA	7.96	3.44	0.02
MARK III	7.95	5.25	0.04

FIGURE 15b. Fission Spectrum Neutron Source and Detector.

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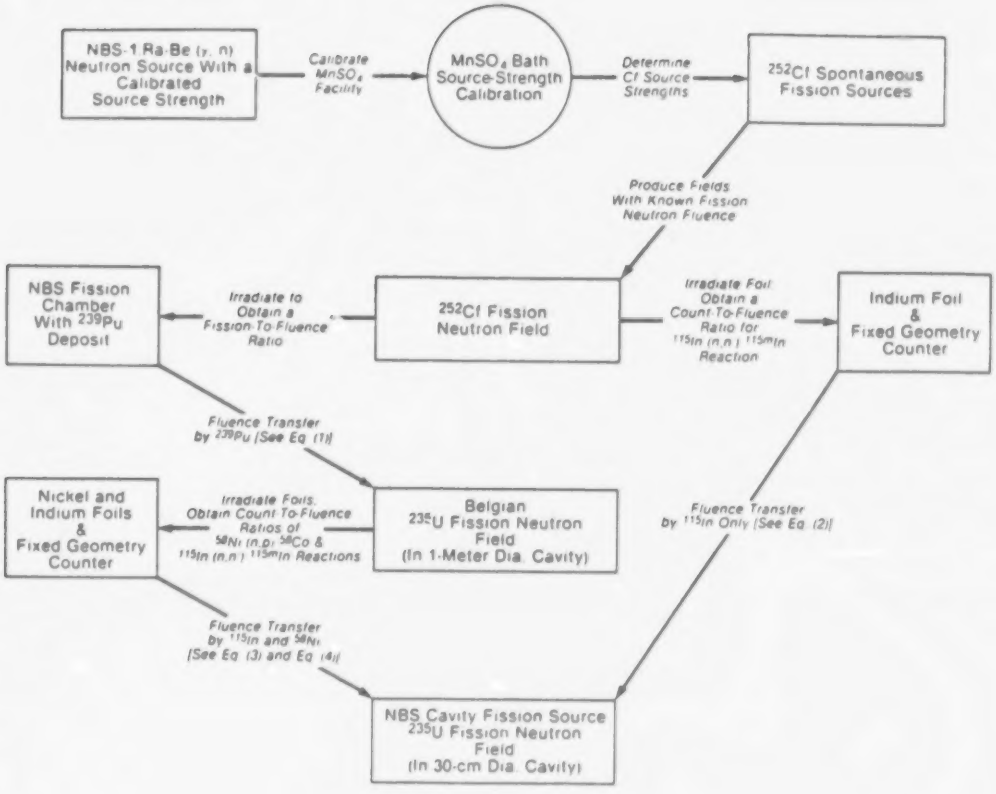
NBS-1 Ra-Be (γ, n) Neutron Source With a Calibrated Source Strength
Calibrate $MnSO_4$ Facility
$MnSO_4$ Bath Source-Strength Calibration
Determine Cf Source Strengths
^{252}Cf Spontaneous Fission Sources
Produce Fields With Known Fission Neutron Fluence
NBS Fission Chamber With ^{239}Pu Deposit
Irradiate to Obtain a Fission-To-Fluence Ratio
^{252}Cf Fission Neutron Field
Irradiate Foil Obtain a Count-To-Fluence Ratio for ^{115}In (n,n') ^{115m}In Reaction
Indium Foil & Fixed Geometry Counter
Fluence Transfer by ^{239}Pu (See Eq. (1))
Nickel and Indium Foils & Fixed Geometry Counter
Irradiate Foils, Obtain Count-To-Fluence Ratios of ^{58}Ni (n,p) ^{58}Co & ^{115}In (n,n') ^{115m}In Reactions
Belgian ^{235}U Fission Neutron Field (In 1-Meter Dia. Cavity)
Fluence Transfer by ^{115}In Only (See Eq. (2))
Fluence Transfer by ^{115}In and ^{58}Ni (See Eq. (3) and Eq. (4))
NBS Cavity Fission Source ^{235}U Fission Neutron Field (In 30-cm Dia. Cavity)

$$\phi^{SCK/CEN}_{U\text{-}235} = \left[\frac{\phi_{cf}}{F^{Pu\text{-}239}_{cf}}\right] \bullet F^{Pu\text{-}239}_{U\text{-}235} \bullet \left[\frac{\bar{\sigma}^{Pu\text{-}239}_{cf}}{\bar{\sigma}^{Pu\text{-}239}_{U\text{-}235}}\right] \qquad (1)$$

$$\text{Old}\ \phi^{NBS}_{U\text{-}235} = \left[\frac{\phi_{cf}}{R^{In\text{-}115}_{cf}}\right] \bullet R^{In\text{-}115}_{U\text{-}235} \bullet \left[\frac{\bar{\sigma}^{In\text{-}115}_{cf}}{\bar{\sigma}^{In\text{-}115}_{U\text{-}235}}\right] \qquad (2)$$

$$\text{New}\ \phi^{NBS}_{U\text{-}235} = \left[\frac{\phi^{SCK/CEN}_{U\text{-}235}}{R^{Ni\text{-}58}_{U\text{-}235\,(SCK)}}\right] \bullet R^{Ni\text{-}58}_{U\text{-}235\,(NBS)} \qquad (3)$$

$$\text{New}\ \phi^{NBS}_{U\text{-}235} = \left[\frac{\phi^{SCK/CEN}_{U\text{-}235}}{R^{In\text{-}115}_{U\text{-}235\,(SCK)}}\right] \bullet R^{In\text{-}115}_{U\text{-}235\,(NBS)} \qquad (4)$$

FIGURE 16. Present and Former Procedures for Calibration of the Neutron Fluence Rate in the NBS ^{235}U Cavity Fission Source.

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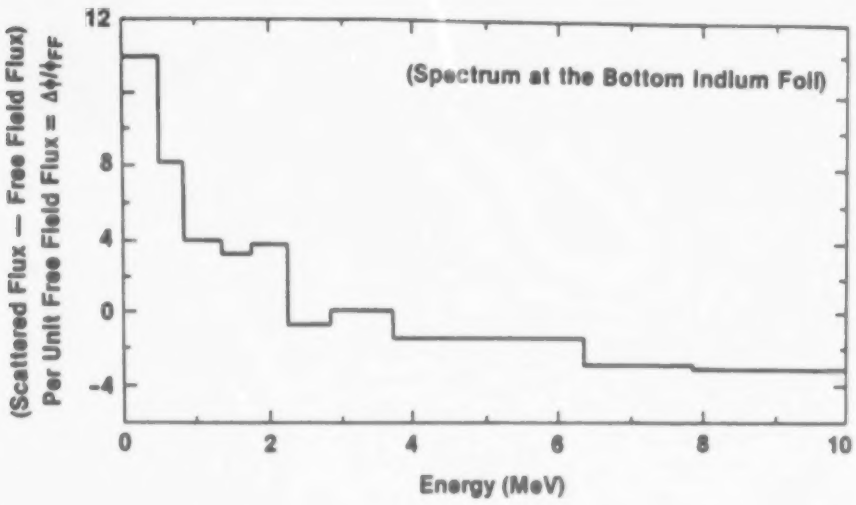

FIGURE 17. Effects of Scattering on the Neutron Energy Spectrum in the NBS Cavity Fission Source.

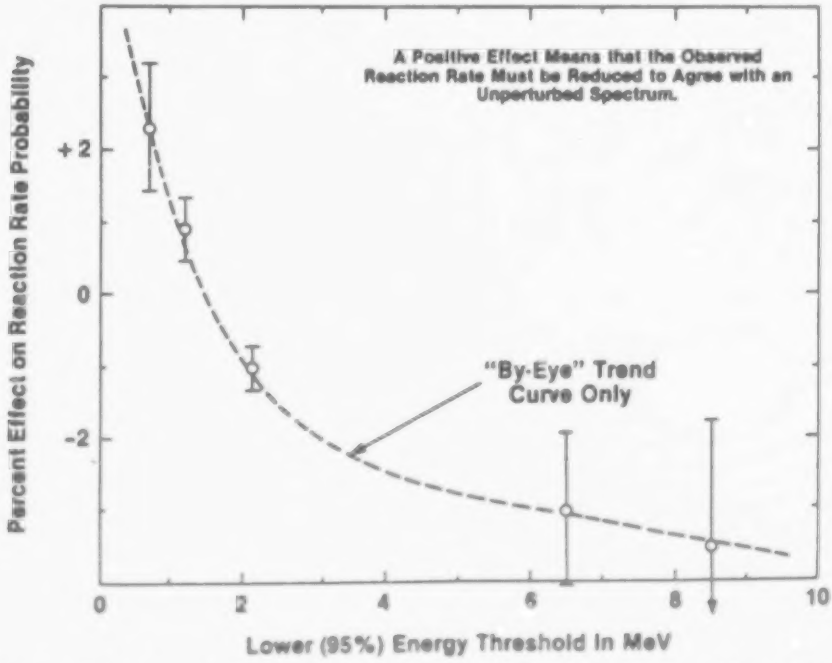

FIGURE 18. Monte Carlo Estimates of Trends in Effects on Threshold-Type Nuclear Reactions Caused by Scattering in the Cavity Fission Source.

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TABLE 1

LWR-PV BENCHMARK FIELD FACILITIES*

Benchmark Field Facility	Location	Anticipated Operation Schedule	Main Purpose
$^{252}Cf/^{235}U$	NBS,US	1975-Open	Standard fields for cross-section testing and validation; emphasis is on equivalent fission flux calibrations and RM fluence counting standards.
PCA-PV	ORNL,US	1978-84	Data base for the "PCA Physics-Dosimetry Blind Test": Low-power experimental/calculational benchmark for different LWR-PV configurations; emphasis is on verification of radial neutron exposure gradients and lead factors; i.e., confirmation of radial through-wall fracture toughness and embrittlement predictions.
PSF-PV	ORNL,US	1980-84	Data base for the "PSF Physics-Dosimetry-Metallurgy Blind Test": High-power LWR-PV physics-dosimetry-metallurgical test; emphasis is on high-temperature and high-fluence simulation of PWR environmental conditions and verification of neutron damage gradients; i.e., confirmation of radial through-wall fracture toughness and embrittlement predictions.
PSF-SDMF	ORNL,US	1979-Open	High-power LWR-PV benchmark: Emphasis is on verification of surveillance capsule perturbations; specific RM, SSTR, HAFM, and DM verification tests, and quality assurance evaluations of commercial dosimetry materials and services; i.e., confirmation of the physics-dosimetry methods, procedures, and data recommended for in-situ in- and ex-vessel surveillance programs.
VENUS	CEN/SCK, Mol, Belgium	1982-Open	Low-power LWR-PV core source boundary benchmark: Emphasis is on verification of effects of new and old fuel management schemes and accuracy of azimuthal lead factors; i.e., confirmation of azimuthal PV-wall fracture toughness and embrittlement predictions.
NESDIP	AEEW, Winfrith, UK	1982-Open	Low-power LWR-PV cavity benchmark: Emphasis is on different PWR configurations and verification, via cavity measurements, of neutron exposure gradients and lead factors; i.e., confirmation of radial through-wall fracture toughness and embrittlement predictions.
DOMPAC	CEA, Fontenay, France,	1980-1983	Low-fluence experimental/calculational benchmark for a specific PWR configuration: Emphasis is on verification of surveillance capsule perturbations and PV-wall neutron exposure and damage gradients; i.e, confirmation of radial PV-wall fracture toughness and embrittlement predictions.

*Acronyms:

- AEEW - Atomic Energy Establishment (Winfrith, UK)
- CEA - Commissariat a l' Energie Atomique (France)
- CEN/SCK - Centre d' Etude de l'Energie Nucleaire-Studiecentrum voor Kernernergie (Mol, Belgium)
- DOMPAC - Triton Reactor Thermal Shield and Pressure Vessel Mockup (Fontenay-aux-Roses)
- UK - United Kingdom
- NBS - National Bureau of Standards, US
- PCA-PV - Pool Critical Assembly Physics-Dosimetry Pressure Vessel Mockup (ORNL)
- ORNL - Oak Ridge National Laboratory
- PSF-PV - Oak Ridge Research (ORR) Reactor Pool Side Facility Physics-Dosimetry-Metallurgy Pressure Vessel Mockup
- PSF-SDMF - PSF Simulated Dosimetry Measurement Facility at the ORR
- VENUS - Critical Facility (Mol, Belgium)
- NESDIP - NESTOR Reactor Surveillance Dosimetry Improvement Program Facility (Winfrith, UK)
- PWR - Pressurized Water Reactor

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TABLE 2

POWER REACTORS BEING USED BY LWR-PV-SDIP PARTICIPANTS TO BENCHMARK PHYSICS-DOSIMETRY METHODS, PROCEDURES AND DATA FOR PRESSURE VESSEL AND SUPPORT STRUCTURE SURVEILLANCE*

(Plant name; reactor type/supplier; reactor operator; ex-vessel cavity (C) and in-vessel (V) surveillance positions available)

Energy Range (MeV)	Type of Dosimeter	Dosimetry Reaction	Nuclear One-1 PWR/B&W Arkansas Power & Light		Nuclear One-2 PWR/CE Arkansas Power & Light		Brown's Ferry-3 BWR/GE Tennessee Valley Authority		H.B. Robinson PWR/WEC Carolina Power & Light		Maine Yankee PWR/CE Maine Yankee Atomic Power		Point Beach-2 PWR/WEC Wisconsin Electric Power		McGuire PWR/WEC Duke Power		CR or Da PWR/B&W		Oconee 1,2&3 PWR/B&W Duke Power[1]		BR-3 PWR/WEC Belgium	
			C	V	C	V	C	V	C	V	C	V	C	V	C	V	C	V	C	V	C	V
>1.5	RMs	$^{63}Cu(n,\alpha)^{60}Co$[f]	Y		Y	Y	Y	Y	Y	Y	Y	Y		Y	Y	Y	P	Y	P		Y	Y
		$^{46}Ti(n,p)^{46}Sc$[f]	Y		Y	Y	Y	Y	Y	Y	Y	Y			Y		P	P	PY		Y	Y
		$^{54}Fe(n,p)^{54}Mn$[f]	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y					P	Y	PY	Y	Y	Y
		$^{58}Ni(n,p)^{58}Co$[f]	Y	Y	Y	Y	Y		Y	Y	Y	Y	Y	Y	Y	Y	P	Y	PY	Y	Y	Y
		$^{238}U(n,f)^{140}Ba$-La	Y	(N)	Y	(N)	Y	Y	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	[N]	[N]	(N)	(N)	(N)	(N)
		$^{238}U(n,f)^{103}Ru$	(Y)	Y	(Y)	(Y)	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y	Y	Y
		$^{238}U(n,f)^{95}Zr$-Nb	(Y)	Y	(Y)	(Y)	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y	Y	Y
		$^{238}U(n,f)^{137}Cs$	(Y)	Y	(Y)	Y	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y	Y	Y
		$^{232}Th(n,f)^{140}Ba$-La		(N)		(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	[N]	[N]	(N)	(N)	(N)	
		$^{232}Th(n,f)^{95}Zr$-Nb					Y	Y									P	P				
		$^{232}Th(n,f)^{137}Cs$					Y	Y									P	P				
		$^{237}Np(n,f)^{140}Ba$-La	Y	(N)	Y	(N)	Y	Y	(N)	(N)	(N)	(N)	(N)	(N)	(N)	(N)	P	(N)	(N)	(N)		(N)
>0.4	RMs	$^{237}Np(n,f)^{103}Ru$	(Y)	Y	(Y)	(Y)	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y		Y
		$^{237}NP(n,f)^{95}Zr$-Nb	(Y)	Y	(Y)	(Y)	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y		Y
		$^{237}Np(n,f)^{137}Cs$	(Y)	Y	(Y)	Y	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	P	Y		Y
		$^{93}Nb(n,n')^{93m}Nb$								Y	Y	Y	Y	Y	Y	Y	P	Y	P	Y		Y
		$^{59}Co(n,\gamma)^{60}Co$[e]								Y	Y	Y					P	P	P			Y
		$^{109}Ag(n,\gamma)^{110m}Ag$[e]	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y			Y	Y	P	Y	PY	Y	Y	Y
		$^{58}Fe(n,\gamma)^{59}Fe$	Y		Y	Y	Y	Y	Y	Y	Y	Y			Y		P	P	P		Y	Y
5×10^{-7}		$^{45}Sc(n,\gamma)^{46}Sc$[e]	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y			Y		P	Y	PY	Y	Y	Y
to											Y	Y			Y		P	P	PY		Y	Y
0.5[b]		$^{235}U(n,f)^{140}Ba$-La	Y	(N)	Y	(N)	Y	Y	(N)	(N)	(N)	(N)	(N)	(N)	(N)		(N)	(N)	(N)	(N)		(N)
		$^{235}U(n,f)^{103}Ru$	(Y)		(Y)		Y	Y	Y	Y	Y	Y			Y		P	P	P			Y
		$^{235}U(n,f)^{95}Zr$-Nb	(Y)		(Y)		Y	Y	Y	Y	Y	Y			Y		P	P	P			Y
		$^{235}U(n,f)^{137}Cs$	(Y)		(Y)		Y	Y	Y	Y	Y	Y			Y		P	P	P			Y
As Above[d]	SSTRs[g]	$^{238}U(n,f)FP$					Y		Y	Y	Y	Y			Y		P	P	P			
		$^{232}Th(n,f)FP$					Y								Y		P	P	P			
		$^{237}Np(n,f)FP$					Y								Y		P	P	P			
		$^{235}U(n,f)FP$					Y		Y	Y	Y	Y			Y		P	P	P			
		$^{239}Pu(n,f)FP$					Y										P	P	P		Y	
<0.1		$^{6}Li(n,\alpha)$															P	P	P			
		$^{10}B(n,\alpha)$															P	P	P			
>0.3		$^{4}He(n,p)$ as CR-39		N		N		N		N		N	(N)	N		N	P	N	(P)			N
>4	HAFMs[d]	$^{58}Ni(n,He)$ as metal[f]							(Y)	Y	(Y)	Y		Y	(Y)	Y	(P)	P	[Y]		(Y)	(Y)
		$^{27}Al(n,He)$ as metal[f]	N		N		N		N	Y	(N)	Y		Y	N	Y	N	P	N		N	(Y)
		$^{63}Cu(n,He)$ as metal[f]	N		N		N		N	Y	(N)	Y		Y	N	Y	N	P	N		N	(Y)
		$^{56}Fe(n,He)$ as metal[f,h]	N		N		N		N	Y	(N)	Y		Y	N	Y	N	P	N		N	(Y)
		$^{7}Li(n,He)$ as LiF																				
		$^{9}Be(n,He)$ as metal								Y	Y						P	P				
>0.1[b]		$^{3}S(n,He)$ as PbS									Y						P	P				(Y)
		$^{19}F(n,He)$ as PbF_2									Y						P	P				
		$^{40}Ca(n,He)$ as CaF_2									Y						P	P				
		$^{58}Ni(n,He)$ as NiO or TiN									Y						P	P				
		$^{35}Cl(n,He)$ as $PbCl_2$									Y						P	P				(Y)
		$^{16}O(n,He)$ as GeO_2															P	P				
<0.1		$^{6}Li(n,He)$ as LiF or alloy								Y	Y						P	P				(Y)
		$^{10}B(n,He)$ nat. or alloy								Y	Y						P	P				(Y)
		Quartz															P	P				
		Sapphire								Y							P	P				
<0.1	DMs[c]	$Al_2O_3B_{f,b}$								Y				Y			P	P				
		$A533B^{f,k}$								Y				(Y)		Y	P	P				
		Other Steel[f]								Y		Y				Y	P	P				P

*See footnotes for this table on next page.

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FOOTNOTES* for Table 2:

Power Reactors Being Used by LWR-PV-SDIP Participants

[a]Energy ranges for the solid state track recorders (SSTRs) are the same as those given for the fissionable radiometric sensors.

[b]Generally these reactions are used with cadmium, cadmium-oxide or gadolinium filters to eliminate their sensitivity to neutrons having energies less than 0.5 eV. The cavity measurements in the Arkansas Power & Light reactors have also included intermediate-energy measurements using thick (1.65 g/cm^2) boron-10 filters (shells) for the ^{235}U, ^{238}U and ^{237}NP fission sensors.

[c]DM means damage monitors (damage to the sensor crystal lattice, such as A302P and A533B or other steels with high copper content and high sensitivity to damage).

[d]HAFM means helium accumulation fluence monitors.

[e]Generally cobalt and silver are included as dilute alloys with aluminum. Scandium is normally Sc_2O_3, and more recently as a ~0.1% Sc_2O_3-MgO ceramic wire.

[f]Frequently when there is no specific HAFM dosimetry package, some of the radiometric sensors and some of the steel damage monitors serve as HAFMs after they have been analyzed for their principal function.

[g]Ni and/or Fe gradient disks were also included in the SSTR capsule, as required.

[h]Iron from RM sensors or Charpy specimens.

[i]Note that power plant CR is Crystal River-3 (Florida Power Corp.) and DB is Davis Besse-1 (Toledo Edison Co.).

[j]The Y following the P refers to a previous Oconee 2 test.

[k]Surveillance capsule reference correlation material (ASTM reference steel plates).

[l]The determination (or feasibility) of using any of the Oconee plants for future benchmark studies has yet to be made.

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TABLE 3

INTEGRAL I- AND J-REACTION RATES (PER WATT OF PCA POWER) FOR THE 1/4-T LOCATION OF THE 12/13 CONFIGURATION IN THE PCA

Reaction	Energy (MeV)	Calculation	Experiment	C/E
I (protons/MeV•s)	0.4467	3.23 (-20)	4.58 (-20)	0.71
I	0.5198	2.64 (-20)	4.18 (-20)	0.63
I	0.5877	2.08 (-20)	3.63 (-20)	0.57
I	0.6515	1.66 (-20)	2.96 (-20)	0.56
I	0.7119	1.26 (-20)	2.19 (-20)	0.58
J (protons/s)	0.4073	1.26 (-20)	2.15 (-20)	0.58
J	0.4837	1.02 (-20)	1.78 (-20)	0.57
J	0.5540	0.846 (-20)	1.47 (-20)	0.58
J	0.6197	0.717 (-20)	1.27 (-20)	0.56

TABLE 4

INFINITE MEDIUM DOSE RATES* OBSERVED IN THE 1981 PCA EXPERIMENTS

	Configuration		
Location	4/12 SSC	12/13	4/12
1/4 T	220	152	490***
1/2 T	65.4	35.6	---
3/4 T	19.1	9.24	---
VB	11.0**	2.56	---

*Dose rates in mrad/h at 1-watt PCA power were corrected for Janus probe field perturbation.

**A perturbation factor of 0.9 has been applied corresponding to that obtained at the VB location in the 12/13 configuration.

***A perturbation factor of 1.16 has been applied corresponding to that obtained at the 1/4-T location in the 4/12 SCC configuration.

TABLE 5

GAMMA-RAY DOSE RATES* FOR THE 4/12 SSC CONFIGURATION

	Experiment		Calculation		
Location	TLD	$\dot{D}_{IM}$	CEN/SCK	$\dot{D}_{IM}$/TLD	$\dot{D}_{IM}$/CAL
1/4 T	255	220	210	0.86	1.05
1/2 T	68	65.4	52	0.96	1.26
3/4 T	21.5	19.1	19.1	0.89	1.00
VB	11.5	11.0	2.2	0.96	5.05

*Dose rates in mrad/h at 1-watt PCA power.

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TABLE 6

COMPARISON BETWEEN EXPERIMENTALLY DETERMINED CHARPY SHIFT AND BLIND TEST PREDICTIONS

	Determined From Charpy Curves			Smallest and Largest Values Predicted by Blind Test Participants		Difference Blind Test - CV81	
	CV81* (°C)	Std. (°C)	MEA** (°C)	Min. (°C)	Max. (°C)	Min. (°C)	Max. (°C)
A302-B Plate							
SSC-1	78	±12	82	71	98	-7	+20
SSC-2	94	±11	94	75	112	-19	+18
0-T	77	±10	81	71	96	-6	+19
1/4 T	65	±18	67	65	81	0	+16
1/2 T	52	±10	50	45	66	-7	+14
A533-B Plate							
SSC-1	71	±11	61	45	69	-24	-2
SSC-2	84	±10	81	62	99	-22	+15
0-T	71	±13	75	60	87	-11	+16
1/4 T	69	±9	69	54	63	-15	-6
1/2 T	52	±10	53	26	52	-26	0
A508 22NiMoCr37 Forging							
SSC-1	52	±16	61	57	77	-5	+25
SSC-2	109	±14	94	65	110	-44	+1
0-T	81	±16	72	63	97	-18	+16
1/4 T	66	±18	78	52	76	-14	+10
1/2 T	66	±13	56	45	64	-21	-2
A508-3 Forging							
SSC-1	15	±7	20	6	43	-9	+18
SSC-2	39	±7	39	11	53	-28	+14
0-T	27	±7	25	10	49	-17	+22
1/4 T	23	±6	20	8	42	-15	+19
1/2 T	22	±7	14	6	35	-16	+13
A533-B Submerged Arc Weld (EC)							
SSC-1	112	±33	108	99	118	-13	+6
SSC-2	123	±60	119	130	153	-7	+30
0-T	125	±50	124	121	135	-4	+10
1/4 T	96	±18	94	91	115	-5	+19
1/2 T	94	±20	89	63	103	-31	+9
A533-B Submerged Arc Weld (R)							
SSC-1	230	±12	222	218	227	-12	-3
SSC-2	309	±38	289	246	319	-63	+10
0-T	294	±15	286	239	288	-55	-6
1/4 T	270	±25	256	180	218	-90	-52
1/2 T	242	±44	239	143	189	-99	-53

*ORNL Evaluation.
**Evaluation in Ref (Ha84 and Ha84a).

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TABLE 7

LIST OF MATERIALS AND CHEMICAL COMPOSITIONS (wt%)

Material	Heat Code	Supplier	P	Ni	Cu
A302-B (ASTM Reference Plate)	F23	NRL	0.011	0.18	0.20
A533-B (HSST Plate 03)	3PS, 3PT, 3PU	NRL	0.011	0.56	0.12
A508 22NiMoCr37 Forging	K	KFA	0.009	0.96	0.12
A508-3 Forging	MO	MOL	0.008	0.75	0.05
A533-B Submerged Arc Weld (Single Vee Type A533-B Base Plate)	EC	EPRI	0.007	0.64	0.24
A533-B Submerged Arc Weld (Single Vee Type A533-B Base Plate)	R	RR&A	0.009	1.58	0.23

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TABLE 8

SUMMARY OF RADIATION DAMAGE DETERMINATIONS FOR THE CHARPY SPECIMEN

	$\phi t > 1.0$ Mev ($n/cm^2 \cdot 10^{19}$)*	$\phi t > 0.1$ MeV ($n/cm^2 \cdot 10^{19}$)	dpa (10^{-3})	ΔNDT 41J (°C)	Std. Dev. (°C)	ΔNDT 68J (°C)	Std. Dev. (°C)	ΔNDT 0.89 mm (°C)	Std. Dev. (°C)	Upper Shelf Drop (J)	Std. Dev. (J)
A302-B Plate											
SSC-1	2.59	7.46	3.86	78	±12	84	±17	86	±10	23	±8
SSC-2	5.38	15.35	7.96	94	±11	(101)**	±15	92	±10	33	±6
0-T	3.95	11.44	6.06	77	±10	(87)	±20	(77)	±16	25	±7
1/4 T	2.16	8.13	3.70	65	±18	(109)	±83	(72)	±15	29	±6
1/2 T	1.03	5.27	2.11	52	±10	(57)	±17	56	±10	26	±7
A533-B Plate											
SSC-1	2.32	6.61	3.44	71	±11	71	±12	85	±9	28	±11
SSC-2	4.83	13.63	7.11	84	±10	86	±11	91	±7	40	±13
0-T	3.59	10.20	5.46	74	±13	(92)	±25	91	±10	42	±13
1/4 T	1.95	7.13	3.28	69	±9	65	±7	74	±6	40	±13
1/2 T	0.94	4.60	1.87	52	±10	52	±8	57	±7	22	±13
A508 22NiMoCr37 Forging											
SSC-1	1.75	5.69	2.77	(52)	±16	57	±11	78	±7	47	±16
SSC-2	3.64	11.70	5.72	109	±14	114	±10	117	±8	69	±16
0-T	2.71	8.70	4.40	81	±16	90	±10	97	±7	43	±16
1/4 T	1.47	6.11	2.64	66	±18	82	±11	93	±8	48	±16
1/2 T	0.71	3.97	1.51	66	±13	71	±9	74	±8	40	±15
A508-3 Forging											
SSC-1	1.93	6.32	3.07	15	±7	13	±5	18	±5	32	±14
SSC-2	4.02	13.02	6.34	39	±7	38	±5	38	±5	37	±14
0-T	2.95	9.65	4.80	27	±7	24	±5	26	±5	29	±11
1/4 T	1.60	6.84	2.92	23	±6	20	±5	23	±5	7	±14
1/2 T	0.77	4.45	1.67	22	±7	24	±6	21	±6	26	±14
A553-B Submerged Arc Weld (EC)											
SSC-1	1.87	6.11	2.97	112	±33	(166)	±120	(142)	±35	33	±14
SSC-2	3.90	12.59	6.14	123	±60	(157)	-	138	±20	33	±14
0-T	2.88	9.45	4.71	125	±50	-	-	(171)	±66	34	±11
1/4 T	1.60	6.82	2.92	96	±18	-	-	(118)	±22	31	±14
1/2 T	0.77	4.48	1.68	94	±20	(136)	±65	(129)	±42	26	±14
A553-B Submerged Arc Weld (R)											
SSC-1	2.46	7.07	3.66	230	±12	259	±21	260	±17	98	±15
SSC-2	5.13	14.56	7.57	309	±38	-	-	352	±64	123	±15
0-T	3.18	10.97	5.83	294	±15	-	-	(364)	±56	110	±15
1/4 T	2.12	7.95	3.62	270	±25	-	-	(321)	±58	108	±14
1/2 T	1.02	5.21	2.09	242	±44	-	-	270	±22	98	±14

*neutrons/$cm^2 \cdot 10^{19}$.

**Values in parentheses are obtained by extrapolation and may be unreliable.

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TABLE 9

RE-EVALUATED EXPOSURE VALUES AND THEIR UNCERTAINTIES FOR LWR PRESSURE VESSEL SURVEILLANCE CAPSULES

Plant	Unit	Capsule	Service Lab*	Fluence (ϕt > 1 MeV) (n/cm²) Old	Fluence (ϕt > 1 MeV) (n/cm²) New [% (1s)]	New/Old	Fluence (E < 0.414 eV) (n/cm²)	dpa [% (1σ)]	New dpa/ϕt	dpa/s	hpa (appb)†	Exposure** Time (s)
Westinghouse					***							
Conn. Yankee		A	BMI	2.08 E+18	3.16 E+18 (12)	1.53	2.54 E+18 (18)	0.00482 (12)	1.52 E-21	9.06 E-11	6	5.233 E+07
Conn. Yankee		F	BMI	4.04 E+18	6.06 E+18 (24)	1.50	5.43 E+18 (32)	0.00949 (27)	1.56 E-21	1.24 E-10	13	7.651 E+07
Conn. Yankee		H	W	1.79 E+19	2.00 E+19 (24)	1.12	2.33 E+19 (19)	0.0324 (27)	1.62 E-21	1.36 E-10	52	2.390 E+08
San Onofre		A	SwRI	1.20 E+19	2.86 E+19 (22)	2.38	2.05 E+19 (23)	0.0486 (27)	1.70 E-21	8.35 E-10	43	5.824 E+07
San Onofre		D	SwRI	2.36 E+19	5.62 E+19 (26)	2.38	3.76 E+19 (23)	0.0944 (29)	1.68 E-21	1.06 E-09	80	8.881 E+07
San Onofre		F	W	5.14 E+19	5.73 E+19 (14)	1.11	2.99 E+19 (28)	0.0955 (20)	1.67 E-21	3.92 E-10	73	2.438 E+08
Turkey Point	3	S	SwRI	1.41 E+19	1.62 E+19 (24)	1.15	1.34 E+19 (24)	0.0255 (27)	1.57 E-21	2.33 E-10	33	1.095 E+08
Turkey Point	3	T	W	5.68 E+18	7.01 E+18 (10)	1.23	5.12 E+18 (58)	0.0109 (12)	1.55 E-21	4.73 E-10	14	2.302 E+07
Turkey Point	4	S	SwRI	1.25 E+19	1.31 E+19 (25)	1.05	1.31 E+19 (25)	0.0213 (27)	1.63 E-21	1.97 E-10	37	1.079 E+08
Turkey Point	4	T	SwRI	6.05 E+18	7.54 E+18 (13)	1.25	8.40 E+18 (21)	0.0130 (13)	1.72 E-21	3.48 E-10	20	3.728 E+07
H. B. Robinson	2	S	W	3.02 E+18	3.91 E+18 (24)	1.29	8.81 E+18 (18)	0.00615 (27)	1.57 E-21	1.06 E-10	19	4.209 E+07
H. B. Robinson	2	V	SwRI	4.51 E+18	7.42 E+18 (22)	1.61	8.96 E+18 (20)	0.0119 (25)	1.59 E-21	1.09 E-10	21	1.050 E+08
Surry	1	T	BMI	2.50 E+18	2.86 E+18 (9)	1.14	3.57 E+18 (20)	0.00449 (12)	1.57 E-21	1.33 E-10	8	3.378 E+07
Surry	2	X	BMI	3.02 E+18	3.03 E+18 (11)	1.00	3.64 E+18 (20)	0.00473 (13)	1.56 E-21	1.28 E-10	9	3.687 E+07
North Anna	1	V	B&W	2.49 E+18	2.72 E+18 (9)	1.09	5.80 E+18 (14)	0.00411 (11)	1.51 E-21	1.15 E-10	11	3.570 E+07
Beznau	2	R	EIFR	1.70 E+19	1.34 E+19 (9)	1.27	2.27 E+19 (21)	0.0198 (11)	1.48 E-21	1.16 E-10	49	1.714 E+08
Pr. Island	1	V	W	5.21 E+18	6.03 E+18 (11)	1.16	9.21 E+18 (21)	0.0102 (16)	1.69 E-21	2.41 E-10	20	4.248 E+07
Pr. Island	2	V	W	5.49 E+18	6.74 E+18 (10)	1.23	9.75 E+18 (26)	0.0117 (13)	1.74 E-21	2.67 E-10	21	4.394 E+07
R. E. Ginna	1	R	W	7.60 E+18	1.17 E+19 (10)	1.54	1.84 E+19 (25)	0.0215 (14)	1.83 E-21	2.59 E-10	38	8.328 E+07
R. E. Ginna	1	V	W	4.90 E+18	5.93 E+18 (14)	1.21	1.37 E+19 (59)	0.0102 (22)	1.72 E-21	2.20 E-10	29	4.612 E+07
Kewaunee		V	W	5.59 E+18	6.41 E+18 (10)	1.15	1.23 E+19 (23)	0.0114 (13)	1.78 E-21	2.82 E-10	26	4.057 E+07
Point Beach	1	S	W	7.05 E+18	8.45 E+18 (10)	1.20	1.20 E+19 (19)	0.0146 (13)	1.73 E-21	1.25 E-10	27	1.163 E+08
Point Beach	1	R	W	2.22 E+19	2.29 E+19 (10)	1.37	2.85 E+19 (22)	0.0408 (13)	1.78 E-21	2.50 E-10	61	1.632 E+08
Point Beach	2	V	BMI	4.74 E+18	7.28 E+18 (11)	1.54	1.09 E+19 (18)	0.0121 (13)	1.66 E-21	2.52 E-10	23	4.805 E+07
Point Beach	2	T	W	9.45 E+18	9.40 E+18 (10)	0.99	1.48 E+19 (21)	0.0157 (12)	1.67 E-21	1.44 E-10	32	1.087 E+08
Point Beach	2	R	W	2.01 E+19	2.52 E+19 (10)	1.25	4.71 E+19 (26)	0.0460 (14)	1.83 E-21	2.81 E-10	93	1.640 E+08
D. C. Cook	1	T	SwRI	1.80 E+18	2.71 E+18 (22)	1.51	3.26 E+19 (19)	0.00445 (25)	1.64 E-21	1.12 E-10	77	3.991 E+07
Indian Point	2	T	SwRI	2.02 E+18	3.28 E+18 (22)	1.62	4.01 E+18 (44)	0.00537 (27)	1.64 E-21	1.20 E-10	91	4.473 E+07
Indian Point	3	T	W	2.92 E+18	3.23 E+18 (22)	1.11	3.13 E+18 (21)	0.00520 (25)	1.61 E-21	1.23 E-10	74	4.211 E+07
Zion	1	T	BMI	1.80 E+18	3.04 E+18 (10)	1.69	3.17 E+18 (21)	0.00488 (12)	1.61 E-21	1.29 E-10	82	3.789 E+07
Zion	1	U	W	8.92 E+18	1.01 E+19 (10)	1.13	8.87 E+18 (24)	0.0166 (13)	1.64 E-21	1.47 E-10	21	1.123 E+08
Zion	2	U	BMI	2.00 E+18	2.80 E+18 (9)	1.40	3.80 E+18 (15)	0.00446 (12)	1.59 E-21	1.11 E-10	10	4.007 E+07
Salem	1	T	W	2.56 E+18	2.84 E+18 (22)	1.11	3.26 E+18 (19)	0.00460 (25)	1.62 E-21	1.34 E-10	7	3.426 E+07

*BMI = Battelle Memorial Institute; W = Westinghouse; SwRI = Southwest Research Institute; CE = Combustion Engineering; ET = Effects Technology; B&W = Babcock and Wilcox, EIFR Eidg. Institute für Reaktorforschung.

**Equivalent constant power level exposure time.

***3.16 E+18 (12) means 3.16×10^{18} with a 12% (1σ) uncertainty.

†Calculated for A3028 steel with a nominal concentration of 0.55 appm boron present.

TABLE 9 (Cont'd)

Plant	Unit	Capsule	Service Lab*	Fluence (ϕt > 1 MeV) (n/cm²) Old	Fluence (ϕt > 1 MeV) (n/cm²) New [% (1s)]	New/Old	Fluence (E < 0.414 eV) (n/cm²)	dpa [% (1σ)]	New dpa/ϕt	dpa/s	hpa (appb)†	Exposure** Time (s)
Combustion Engineering												
Palisades		A240	BMI	4.40 E+19	6.06 E+19 (23)	1.38	7.26 E+19 (61)	0.0972 (28)	1.60 E-21	1.36 E-09	170	7.130 E+07
Fort Calhoun		W225	CE	5.10 E+18	5.83 E+18 (14)	1.14	3.09 E+19 (60)	0.00879 (18)	1.51 E-21	1.07 E-10	63	8.191 E+07
Maine Yankee		1	ET	1.30 E+19	1.76 E+19 (19)	1.35	3.00 E+19 (29)	0.0285 (23)	1.62 E-21	1.03 E-09	62	2.777 E+07
Maine Yankee		2	W	8.84 E+19	7.73 E+19 (13)	0.87	1.20 E+20 (23)	0.121 (18)	1.57 E-21	8.38 E-10	230	1.446 E+08
Maine Yankee		W263	BMI	7.10 E+18	5.67 E+18 (12)	0.82	2.67 E+19 (21)	0.00843 (14)	1.49 E-21	5.83 E-11	55	1.446 E+08
Babcock & Wilcox												
Oconee	1	F	B&W	8.70 E+17	6.98 E+17 (21)	0.80	1.00 E+18 (13)	0.000959 (19)	1.37 E-21	3.65 E-11	3	2.629 E+07
Oconee	1	E	B&W	1.50 E+18	1.50 E+18 (10)	1.00	2.61 E+18 (15)	0.00208 (10)	1.39 E-21	4.01 E-11	7	5.186 E+07
Oconee	2	C	B&W	9.43 E+17	1.01 E+18 (10)	1.07	1.55 E+18 (15)	0.00148 (11)	1.47 E-21	3.88 E-11	4	3.802 E+07
Oconee	3	A	B&W	7.39 E+17	8.05 E+17 (10)	1.09	1.34 E+18 (11)	0.00113 (11)	1.40 E-21	3.79 E-11	3	2.983 E+07
Three Mile Is.	1	E	B&W	1.07 E+18	1.09 E+18 (9)	1.02	1.90 E+18 (11)	0.00151 (9)	1.39 E-21	3.75 E-11	5	4.036 E+07
Arkansas Nuclear	1	E	B&W	7.27 E+17	8.18 E+17 (8)	1.13	6.32 E+17 (9)	0.00117 (8)	1.43 E-21	3.92 E-11	2	2.981 E+07
General Electric												
Dresden	3	4G14	W	2.06 E+19	1.86 E+19 (17)		1.51 E+20 (62)	0.0285 (17)	1.53 E-21	3.35 E-10	290	8.483 E+07
		4G15		1.50 E+19	1.35 E+19 (17)		1.19 E+20 (62)	0.0209 (17)	1.55 E-21	2.46 E-10	240	8.483 E+07
		4G16		1.20 E+19	1.08 E+19 (17)	0.89	9.70 E+19 (62)	0.0168 (17)	1.55 E-21	1.98 E-10	200	8.483 E-07
		4G17		5.16 E+18	4.51 E+18 (17)		5.20 E+19 (62)	0.00733 (18)	1.63 E-21	8.64 E-11	120	8.438 E+07
Quad Cities	2	3G14	W	4.14 E+19	4.28 E+19 (16)		2.45 E+20 (62)	0.0611 (17)	1.43 E-21	4.29 E-10	400	1.422 E+08
		3G15		3.48 E+19	3.60 E+19 (16)		2.13 E+20 (62)	0.0516 (17)	1.43 E-21	3.63 E-10	370	1.422 E+08
		3G16		2.43 E+19	2.52 E+19 (16)	1.03	1.55 E+20 (62)	0.0362 (17)	1.44 E-21	2.54 E-10	290	1.422 E+08
		3G17		2.32 E+19	2.37 E+19 (17)		1.49 E+20 (62)	0.0342 (17)	1.44 E-21	2.41 E-10	290	1.422 E+08
Quad Cities	1	3G5	W	4.04 E+19	4.23 E+19 (17)		2.41 E+20 (62)	0.0604 (17)	1.43 E-21	4.85 E-10	400	1.243 E+08
		3G6		3.08 E+19	3.12 E+19 (17)		1.92 E+20 (62)	0.0450 (17)	1.44 E-21	3.62 E-10	340	1.243 E+08
		3G7		2.37 E+19	2.47 E+19 (17)	1.01	1.54 E+20 (62)	0.0356 (17)	1.44 E-21	2.86 E-10	290	1.253 E+08
		3G8		1.24 E+19	1.17 E+19 (17)		1.03 E+20 (62)	0.0180 (17)	1.54 E-21	1.45 E-10	210	1.243 E+08
					Avg	1.25						

*BMI = Battelle Memorial Institute; W = Westinghouse; SwRI = Southwest Research Institute; CE = Combustion Engineering; ET = Effects Technology; B&W = Babcock and Wilcox, EIFR Eidg. Institute für Reaktorforschung.
**Equivalent constant power level exposure time.
***3.16 E+18 (12) means 3.16×10^{18} with a 12% (1σ) uncertainty.
†Calculated for A302B steel with a nominal concentration of 0.55 appm boron present.

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TABLE 10

COMPARISON OF HEDL TREND CURVE FORMULA CALCULATIONS OF CHARPY 41-J TEMPERATURE SHIFT AND VALUES DETERMINED BY HAWTHORNE (°C)

	F		3P		R		EC		MO		K	
Position	JRH*	CALC**	JRH	CALC	JRH	CALC	JRH	CALC	JRH	CALC	JRH	CALC
SSC-1	82	69.8	61	54.9	222	232.2	108	115.0	20	8.6	61	54
SSC-2	94	78.1	81	61.8	289	266.2	119	134.2	39	9.8	94	61.9
OT	81	74.9	75	59.0	286	253.8	114	129.7	25	9.5	72	59.7
1/4 T	67	69.2	69	54.2	256	232.2	94	117	20	8.7	78	54.1
1/2 T	50	61.8	53	48.1	239	205.5	89	102.2	14	7.6	56	47.3

*Hawthorne, Ref (Ha84,Ha84a).
**Guthrie, Ref (Gu84b).

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TABLE 11

INTEGRAL I- AND J-REACTION RATES* FOR THE V1 (CORE CENTER) LOCATION IN VENUS

REACTION (ENERGY, MeV)		CALCULATION	EXPERIMENT	C/E
I**	(0.4467)	6.35 (-15)	5.07 (-15)	1.25
I	(0.5198)	5.00 (-15)	4.62 (-15)	1.08
I	(0.5877)	4.05 (-15)	4.10 (-15)	0.99
I	(0.6515)	3.50 (-15)	3.55 (-15)	0.99
I	(0.7119)	3.02 (-15)	3.01 (-15)	1.01
J***	(0.4073)	3.04 (-15)	3.04 (-15)	1.00
J	(0.4837)	2.59 (-15)	2.65 (-15)	0.98
J	(0.5540)	2.25 (-15)	2.33 (-15)	0.97
J	(0.6197)	1.995 (-15)	2.06 (-15)	0.97

* REACTION RATES NORMALIZED TO 100% POWER
** UNITS OF PROTONS/(MeV·S).
*** UNITS OF PROTONS/S.

TABLE 12

INTEGRAL I- AND J-REACTION RATES* FOR THE 24° LOCATION OF THE OUTER BAFFLE IN VENUS

REACTION (ENERGY, MeV)		CALCULATION	EXPERIMENT	C/E
I**	(0.4467)	2.20 (-15)	1.76 (-15)	1.25
I	(0.5198)	1.70 (-15)	1.13 (-15)	1.50
I	(0.5877)	1.35 (-15)	0.870 (-15)	1.55
I	(0.6515)	1.16 (-15)	0.783 (-15)	1.48
I	(0.7119)	0.987 (-15)	0.711 (-15)	1.39
J***	(0.4073)	0.964 (-15)	0.671 (-15)	1.44
J	(0.4837)	0.806 (-15)	0.537 (-15)	1.50
J	(0.5540)	0.693 (-15)	0.458 (-15)	1.51
J	(0.6197)	0.607 (-15)	0.401 (-15)	1.52

* REACTION RATES NORMALIZED TO 100% POWER
** UNITS OF PROTONS/(MeV·S).
*** UNITS OF PROTONS/S.

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TABLE 13

INTEGRAL I- AND J-REACTION RATES* FOR THE 42° LOCATION OF THE OUTER BAFFLE IN VENUS

REACTION (ENERGY, MeV)		CALCULATION	EXPERIMENT	C/E
I**	(0.4467)	6.98 (-15)	4.31 (-15)	1.62
I	(0.5198)	5.43 (-15)	3.95 (-15)	1.37
I	(0.5877)	4.32 (-15)	3.54 (-15)	1.22
I	(0.6515)	3.71 (-15)	3.09 (-15)	1.19
I	(0.7119)	3.17 (-15)	2.64 (-15)	1.20
J***	(0.4073)	3.09 (-15)	2.56 (-15)	1.21
J	(0.4837)	2.59 (-15)	2.23 (-15)	1.16
J	(0.5540)	2.23 (-15)	1.95 (-15)	1.14
J	(0.6197)	1.95 (-15)	1.72 (-15)	1.14

* REACTION RATES NORMALIZED TO 100% POWER
** UNITS OF PROTONS/(MeV·S).
*** UNITS OF PROTONS/S.

TABLE 14

INTEGRAL I- AND J-REACTION RATES* FOR THE 21° LOCATION OF THE BARREL IN VENUS

REACTION (ENERGY, MeV)		CALCULATION	EXPERIMENT	C/E
I**	(0.4467)	5.13 (-16)	3.46 (-16)	1.48
I	(0.5198)	3.91 (-16)	2.66 (-16)	1.47
I	(0.5877)	3.08 (-16)	2.05 (-16)	1.51
I	(0.6515)	2.62 (-16)	1.63 (-16)	1.61
I	(0.7119)	2.23 (-16)	1.38 (-16)	1.62
J***	(0.4073)	2.18 (-16)	1.76 (-16)	1.24
J	(0.4837)	1.82 (-16)	1.37 (-16)	1.33
J	(0.5540)	1.56 (-16)	1.18 (-16)	1.32
J	(0.6197)	1.36 (-16)	1.05 (-16)	1.30

* REACTION RATES NORMALIZED TO 100% POWER
** UNITS OF PROTONS/(MeV·S).
*** UNITS OF PROTONS/S.

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TABLE 15

INTEGRAL I- AND J-REACTION RATES* FOR THE 41° LOCATION OF THE BARREL IN VENUS

REACTION (ENERGY, MeV)		CALCULATION	EXPERIMENT	C/E
I**	(0.4467)	2.39 (-16)	1.58 (-16)	1.51
I	(0.5198)	1.84 (-16)	1.43 (-16)	1.29
I	(0.5877)	1.47 (-16)	1.25 (-16)	1.17
I	(0.6515)	1.26 (-16)	1.08 (-16)	1.16
I	(0.7119)	1.08 (-16)	0.927 (-16)	1.16
J***	(0.4073)	1.08 (-16)	1.07 (-16)	1.01
J	(0.4837)	0.912 (-16)	0.949 (-16)	0.96
J	(0.5540)	0.789 (-16)	0.848 (-16)	0.93
J	(0.6197)	0.696 (-16)	0.766 (-16)	0.91

* REACTION RATES NORMALIZED TO 100% POWER
** UNITS OF PROTONS/(MeV·S).
*** UNITS OF PROTONS/S.

TABLE 16

INTEGRAL I- AND J-REACTION RATES* FOR THE 21° LOCATION OF THE PAD IN VENUS

REACTION (ENERGY, MeV)		CALCULATION	EXPERIMENT	C/E
I**	(0.4467)	5.07 (-17)	4.31 (-17)	1.18
I	(0.5198)	3.85 (-17)	3.95 (-17)	0.98
I	(0.5877)	3.00 (-17)	3.50 (-17)	0.86
I	(0.6515)	2.53 (-17)	2.92 (-17)	0.87
I	(0.7119)	2.13 (-17)	2.31 (-17)	0.93
J***	(0.4073)	2.07 (-17)	2.26 (-17)	0.92
J	(0.4837)	1.71 (-17)	1.93 (-17)	0.88
J	(0.5540)	1.45 (-17)	1.65 (-17)	0.88
J	(0.6197)	1.26 (-17)	1.42 (-17)	0.88

* REACTION RATES NORMALIZED TO 100% POWER
** UNITS OF PROTONS/(MeV·S).
*** UNITS OF PROTONS/S.

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TABLE 17

RECAPITULATION OF THE CERTIFIED ^{235}U FLUENCE STANDARDS ISSUED BY NBS IN FY83 AND FY84 TO BENCHMARK REFERENCE RADIOMETRIC DOSIMETRY FOR THE LWR-PV-SDIP

No	Irrad. Ident.	End of Irrad.	Exposure Time (hr)	Standard I.D.	Prepared for:	Approx date supplied
1	Fx-1-10/82	10/4/82	69.4	Ni-A	(1) Williams/Cogburn; Univ. of Arkansas	Nov 82
2	"	"	"	Fe-FN		
3	"	"	"	U28-FX-1	(2) Failey/Lowry; Battele-Columbus	
4	Ti/Ni-1	1/25/83	96.9	Ni-B*	Kellogg; HEDL	Mar 83
5	"	"	"	Ni-U	Baldwin/Kam; ORNL	Aug 83
6	"	"	"	Ti-11	Univ. of Ark & Battelle-Columbus	Mar 83
7	"	"	"	Ti-14	Kellogg; HEDL	Mar 83
8	Fe/Ni-1	2/6/83	72.3	Fe-GA	Schneider; KFA, Julich	Dec 83
9	"	"	"	Fe-GB	Kellogg: HEDL	Mar 83
10	"	"	"	Ni-P	Schneider; KFA, Julich	Dec 83
11	"	"	"	Ni-R	Kellogg; HEDL	Jan 84
12	Ti/Fe-1	3/1/83	95.0	Fe-GC	Tourwe; Mol	May 83
13	"	"	"	Fe-GD	Kellogg; HEDL	Jan 84
14	"	"	"	Ti-15	Tourwe; Mol	May 83
15	"	"	"	Ti-16	Baldwin/Kam; ORNL	Aug 83
16	"	"	"	Ti-17	(NBS: Internal Use)	
17	"	"	"	Ni-S	Tourwe; Mol	May 83
18	Rh/Al-1	4/6/83	2.0	Rh-I	All for Kellogg; HEDL (On site, at NBS, calibration of the portable Ge-Li gamma counter used at VENUS and NESDIP)	Apr 83
19	"	"	"	Rh-II		
20	"	"	"	Rh-III		
21	"	"	"	Al-11-1		
22	"	"	"	Al-11-2		
23	Al/Fe/Ti-1	4/8/83	16/8	Fe-FE		Apr 83
24	"	"	"	Fe-FC		
25	"	"	"	Al-11-3		
26	"	"	"	Al-11-4		
27	"	"	"	Ti-7		

*Also Ni-BG from standard irradiation which ended on 12/6/81

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TABLE 17 (Cont'd)

No.	Irrad. Ident.	End of Irrad.	Exposure Time(hr)	Standard I.D.	Prepared for:	Approx date supplied
28	Ti/Fe-2	8/26/83	94.2	Fe/Ni-A	Rogers; EG&G-Idaho	Jun 84
29	"	"	"	Ti-46A	Tourwe; Mol: special enriched Ti-46	Sep 83
30	"	"	"	Ti-46B		
31	"	"	"	Ti-A	(NBS: Internal Use)	
32	U-Fe-1	11/14/83	94.6	Fe/Ni-B	Schneider; KFA-Julich	Dec 83
33	"	"	"	Ti-B	Rogers; EG&G-Idaho	Jun 84
34	"	"	"	Ni-C	Rogers; EG&G-Idaho	Jun 84
35	"	"	"	Ni-D	(NBS: Internal Use)	
36	"	"	"	U8-5-21	Schneider; KFA-Julich	Dec 83
37	U/Fe-2	11/20/83	97.8	Ni-E	(NBS: Internal Use)	
38	"	"	"	Ti-C	(NBS: Internal Use)	
39	"	"	"	Fe/Ni-C	Kellogg; HEDL	Jan 84
40	"	"	"	Fe/Ni-D	(NBS: Internal Use)	
41	"	"	"	U8-5-27	Kellogg; HEDL	Jan 84
42	Ti-Fe-3	8/6/84	95.0	Ti-2	Blackburn/Anderson; WEC-Nucl. Tech. Div.	Jan 85
43	"	"	"	Ti-3	Available	
44	"	"	"	Fe-GH	Blackburn/Anderson; WEC-Nucl. Tech. Div.	Jan 85
45	"	"	"	Fe-GK	Available	
46	"	"	"	Ni-G	Blackburn/Anderson; WEC-Nucl. Tech. Div.	Jan 85
47	U/ORNL	8/15/84	1.7	Cu-1	NBS/HEDL	
48	"	"	"	Cu-2	Calibration of 1984	Aug 84
49	"	"	"	U8-6	PCA Irradiations	
50	U/Fe-3	8/30/84	149	U8(Nat)-50	NBS: Cs-137 Flu. Std.	
51	"	"	"	U8(Nat)-51	Rogers; EG&G-Idaho	
52	"	"	"	U8(Nat)-52	Matsamoto; HEDL	
53	"	"	"	NB-1	Williamson; Univ. of Vir.	Sep 84
54	"	"	"	Fe/Ni-150	NBS: Iron-nickel alloy std.	
55	"	"	"	Ni-U	Schima; NBS	Nov 84
56	"	"	"	Ni-T	Fabry; Mol	Oct 84

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TABLE 18

DOSIMETER TYPES AND REACTIONS

Fluence Standard			Number
Type	Reaction	Identification	Distributed
Nickel	$^{58}Ni(n,p)^{58}Co$	Ni-B, G, P, R, S, T, U	7
Iron	$^{54}Fe(n,p)^{54}Mn$	Fe-GA, GB, GC, GD, GH	5
Titanium	$^{46}Ti(n,p)^{46}Sc$	Ti-2, 14. 15, 16	4
U-238	$^{238}U(n,f)$Ba-La	U8-5-21, 5-27	2*
U-238	$^{238}U(n,f)$Cs-137	U8-Nat-52	1**
Fe/Ni alloy	(both ^{58}Co & ^{54}Mn)	Fe/Ni-B, C	2

*Not all laboratories interested in short half-life Ba-La decay.
**Just distributed; to be sent to all laboratories interested.

TABLE 19

RATIO OF MEASURED REACTION CROSS SECTIONS IN THE ^{235}U AND ^{252}Cf FISSION NEUTRON FIELDS AND COMPARISONS WITH RATIOS CALCULATED FROM ENDF/B-V CROSS SECTIONS AND VARIOUS ^{235}U SPECTRA

Isotope and Reaction	$\frac{\sigma \text{ in } ^{235}U}{\sigma \text{ in } ^{252}Cf}$ Experimental Result	Calculated Ratio†/Experimental Result		
		ENDF/B-V Watt for ^{235}U	Madland-Nix "Exact" for ^{235}U	NBS Evaluation for ^{235}U
^{239}Pu (n,f)	(0.997)	1.002	1.004	1.*
^{235}U (n,f)	0.987 ± 1.9%	1.013	1.012	1.013
^{238}U (n,f)	0.957 ± 2.2%	1.017	1.023	0.982
^{237}Np (n,f)	0.995 ± 2.1%	1.001	1.007	0.983
^{240}Pu (n,f)	0.996 ± 2.7%	1.001	1.007	0.982
^{233}U (n,f)	1.030 ± 3.1%	0.972	0.972	0.975
^{232}Th (n,f)	0.928 ± 2.7%	1.035	1.039	0.999
^{115}In (n,n')	0.969 ± 2.1%	1.016	1.021	0.984
^{58}Ni (n,p)	0.917 ± 2.7%	1.006	0.997	0.968

*Exactly 1. due to flux transfer normalization.

†The NBS evaluation of the ^{252}Cf fission neutron spectrum is used for all the calculated values.

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TABLE 20

FULLY CORRECTED EXPERIMENTAL RESULTS FROM VARIOUS EXPERIMENTAL CONFIGURATIONS

Isotope and Reaction	Nuclear Reactions (nuclei)(monitor) (10^{-21})	Cross Section $\sigma_f(^{239}Pu)$	Cross Section by Flux Transfer (mb)
	MARK IIA CENTER*		
^{239}Pu (n,f)	21.65 + 0.9%	1.0	(1818 ± 1.9%)†
^{235}U (n,f)	14.28 ± 1.0%	0.660 ± 1.1%	1199 ± 1.9%
^{238}U (n,f)	3.727 ± 1.8%	0.172 + 1.8%	313 ± 2.4%
^{237}Np (n,f)	16.26 ± 1.3%	0.751 + 1.4%	1366 ± 2.1%
^{240}Pu (n,f)	15.86 ± 1.3%	0.733 ± 1.4%	1332 ± 2.1%
^{233}U (n,f)	23.21 ± 2.5%	1.072 ± 2.6%	1949 ± 3.1%
^{232}Th (n,f)	0.988 ± 2.6%	0.0456 ± 2.6%	83 ± 3.1%
^{115}In (n,n')	2.258 ± 1.2%	0.1043 ± 1.4%	190 ± 2.1%
^{58}Ni (n,p)	1.325 ± 1.7%	0.0612 ± 1.8%	111 ± 2.4%
	MARK III Center*		
^{239}Pu (n,f)	24.57 ± 1.4%	1.0	(1818 ± 1.9%)†
^{235}U (n,f)	16.04 ± 1.7%	0.653 ± 2.0%	1190 ± 2.6%
^{238}U (n,f)	4.201 ± 1.8%	0.171 ± 2.1%	312 ± 2.6%
^{237}Np (n,f)	18.26 ± 1.3%	0.743 ± 1.8%	1355 ± 2.4%
	MARK II Center*		
^{239}Pu (n,f)	15.64 ± 0.8%	1.0	(1818 ± 1.9%)†
^{235}U (n,f)	10.39 ± 1.0%	0.664 ± 1.0%	1211 ± 1.9%
^{238}U (n,f)	2.675 ± 1.6%	0.171 ± 1.6%	312 ± 2.3%
^{237}Np (n,f)	11.67 ± 1.2%	0.746 ± 1.4%	1360 ± 2.1%

*These results apply to a 0.5-inch (1.27 cm) diameter foil positioned as shown in Figure 1. These results have not been corrected to correspond to a point detector at the facility center.

†This value is taken from NBS measurements in the ^{252}Cf spectrum and a calculated ratio, based on ENDF/B-V cross section data and the NBS evaluation of the ^{252}Cf and ^{235}U neutron spectra.

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NE-53
Washington, DC 20545

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Richland, WA 99352

EP Simonen

Battelle Memorial Institute (2)
505 King Avenue
Columbus, OH 43201

MP Manahan
JS Perrin, ONWI

Bechtel Power Corporation
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This report describes progress made in the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) during FY 1984. The primary concern of this program is to improve, test, verify, and standardize the physics-dosimetry-metallurgy and the associated reactor and damage analysis procedures and data used for predicting the integrated effects of neutron exposure to LWR-PVs and support structures. These procedures and data are being recommended in a new and updated set of ASTM standards being prepared, tested, and verified by program participants. These standards, together with parts of the US Code of Federal Regulations and ASME codes, are needed and used for the assessment and control of the condition of LWR-PVs and support structures during the 30- to 60-year lifetime of a nuclear power plant.

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